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

June 28, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear Operations, Inc.
R. E. Ginna Nuclear Power Plant
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: LICENSE AMENDMENT REQUEST TO RELOCATE SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE CONTROLLED PROGRAM – ADOPTION OF TSTF-425, REVISION 3 (CAC NO. MF6358)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment No. 122 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant (Ginna). This amendment is in response to your application dated June 4, 2015, as supplemented by letters dated February 3, 2016, March 29, 2016, and June 16, 2016. Exelon Generation Company, LLC submitted a license amendment request to change the Technical Specifications (TSs) by relocating specific TS surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force - 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force Initiative 5b". Additionally, the change added a new program, the Surveillance Frequency Control Program, to TS Section 5, Administrative Controls.

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

Sincerely,

Diane Render, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 122 to Renewed License No. DPR-18
2. Safety Evaluation

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R.E. GINNA NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-244

R.E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 122
Renewed License No. DPR-18

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (Exelon, the licensee) dated June 4, 2015, as supplemented by letters dated February 3, 2016, March 29, 2016, and June 16, 2016 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-18 is hereby amended to read as follows:

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 122 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

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

Remove	Insert
1.1-4	1.1-4
3.1.1-1	3.1.1-1
3.1.2-2	3.1.2-2
3.1.4-3	3.1.4-3
3.1.5-1	3.1.5-1
3.1.6-2	3.1.6-2
3.1.8-2	3.1.8-2
3.2.1-3	3.2.1-3
3.2.1-4	3.2.1-4
3.2.2-2	3.2.2-2
3.2.3-1	3.2.3-1
3.2.4-3	3.2.4-3
3.3.1-8	3.3.1-8
3.3.1-9	3.3.1-9
3.3.1-10	3.3.1-10
3.3.2-3	3.3.2-3
3.3.2-4	3.3.2-4
3.3.3-2	3.3.3-2
3.3.4-2	3.3.4-2
3.3.5-3	3.3.5-3
3.3.6-2	3.3.6-2
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
3.4.4-1	3.4.4-1
3.4.5-2	3.4.5-2
3.4.6-2	3.4.6-2
3.4.7-2	3.4.7-2
3.4.8-2	3.4.8-2
3.4.9-1	3.4.9-1
3.4.11-3	3.4.11-3
3.4.12-4	3.4.12-4

Remove	Insert
3.4.12-5	3.4.12-5
3.4.13-2	3.4.13-2
3.4.14-2	3.4.14-2
3.4.14-3	3.4.14-3
3.4.15-3	3.4.15-3
3.4.16-2	3.4.16-2
3.5.1-1	3.5.1-1
3.5.1-2	3.5.1-2
3.5.2-2	3.5.2-2
3.5.2-3	3.5.2-3
3.5.4-1	3.5.4-1
3.6.2-4	3.6.2-4
3.6.3-6	3.6.3-6
3.6.3-7	3.6.3-7
3.6.4-1	3.6.4-1
3.6.5-1	3.6.5-1
3.6.6-2	3.6.6-2
3.6.6-3	3.6.6-3
3.7.2-2	3.7.2-2
3.7.4-1	3.7.4-1
3.7.5-3	3.7.5-3
3.7.6-1	3.7.6-1
3.7.7-2	3.7.7-2
3.7.8-2	3.7.8-2
3.7.9-2	3.7.9-2
3.7.10-1	3.7.10-1
3.7.11-1	3.7.11-1
3.7.12-1	3.7.12-1
3.7.14-1	3.7.14-1
3.8.1-3	3.8.1-3
3.8.1-4	3.8.1-4
3.8.1-5	3.8.1-5
3.8.3-2	3.8.3-2
3.8.4-2	3.8.4-2
3.8.6-1	3.8.6-1
3.8.6-2	3.8.6-2
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
5.5-13	5.5-13

(b) Exelon Generation pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the RG&E's application dated December 14, 1979 (transmitted by letter dated December 20. 1979). as supplemented February 20,1980, and March 5, 1980;

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

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

(5) Exelon Generation 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 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; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state power levels up to a maximum of 1775 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 122 are hereby incorporated in the renewed license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

Exelon Generation shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated March 28, 2013, supplemented by letters dated December 17, 2013; January 29, 2014; February 28, 2014; September 5, 2014; September 24, 2014; December 4, 2014; March 18, 2015; June 11, 2015; August 7, 2015; and as approved in the safety evaluation report dated November 23, 2015. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission (NRC).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with the Low Temperature Overpressurization Protection System for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1775 MWt.
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 rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

	SURVEILLANCE	FREQUENCY
SR 3.1.2.2	- NOTE - 1. Only required after 60 effective full power days (EFPD). 2. The predicted reactivity values must be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. Verify measured core reactivity is within ± 1% $\Delta k/k$ of predicted values.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.2	- NOTE - Only required to be performed if the rod position deviation monitor is inoperable. Verify individual rod positions within alignment limit.	Once within 4 hours and in accordance with the Surveillance Frequency Control Program
SR 3.1.4.3	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core to a MRPI transition in either direction.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.4	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. Both reactor coolant pumps operating.	Once prior to reactor criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limit

LCO 3.1.5 The shutdown bank shall be at or above the insertion limit specified in the COLR.

- NOTE -

The shutdown bank may be outside the limit when required for performance of SR 3.1.4.3.

APPLICABILITY: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shutdown bank not within limit.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown bank to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with K_{eff} < 1.0.	6 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.3	- NOTE - Only required to be performed if the rod insertion limit monitor is inoperable. Verify each control bank insertion is within the limits specified in the COLR.	Once within 4 hours and in accordance with the Surveillance Frequency Control Program
SR 3.1.6.4	Verify each control bank not fully withdrawn from the core is within the sequence and overlap limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a COT on power range and intermediate range channels per SR 3.3.1.7 and SR 3.3.1.8.	Once within 7 days prior to criticality
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 530°F.	In accordance with the Surveillance Frequency Control Program
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

- NOTE -

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

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified <u>AND</u> In accordance with the Surveillance Frequency Control Program

	SURVEILLANCE	FREQUENCY
SR 3.2.1.2	- NOTE - If measurements indicate that the maximum over z $[F_Q^C(Z) / K(Z)]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z $[F_Q^C(Z) / K(Z)]$ has not increased.	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified <u>AND</u> In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program
SR 3.2.2.2	- NOTE - Only required to be performed if one power range channel is inoperable with THERMAL POWER ≥ 75% RTP. Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once within 24 hours and in accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

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

- NOTE -

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

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	- NOTE - 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program
SR 3.2.4.2	- NOTE - Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	In accordance with the Surveillance Frequency Control Program

CONDITION		REQUIRED ACTION		COMPLETION TIME
		W.2	Restore trip mechanism or train to OPERABLE status.	48 hours
X.	Required Action and associated Completion Time of Condition W not met.	X.1	Initiate action to fully insert all rods.	Immediately
		AND		
		X.2	Place the Control Rod Drive System in a Condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	- NOTE - Required to be performed within 12 hours after THERMAL POWER is ≥ 50% RTP. Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output and adjust if calorimetric power is > 2% higher than indicated NIS power.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.3	- NOTE - 1. Required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling and if the Surveillance has not been performed within the last 31 EFPD. 2. Performance of SR 3.3.1.6 satisfies this SR. Compare results of the incore detector measurements to NIS AFD and adjust if absolute difference is ≥ 3%.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.6	- NOTE - Not required to be performed until 7 days after THERMAL POWER is ≥ 50% RTP, but prior to exceeding 90% RTP following each refueling. Calibrate excore channels to agree with incore detector measurements.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.7	- NOTE - Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entering MODE 3. Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.8	- NOTE - 1. Not required for power range and intermediate range instrumentation until 4 hours after reducing power < 6% RTP. 2. Not required for source range instrumentation until 4 hours after reducing power < 5E-11 amps. Perform COT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	- NOTE - Setpoint verification is not required. Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.10	- NOTE - Neutron detectors are excluded. Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.11	Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.12	- NOTE - Setpoint verification is not required. Perform TADOT.	Prior to reactor startup if not performed within previous 31 days
SR 3.3.1.13	Perform COT.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. As required by Required Action A.1 and referenced by Table 3.3.2-1.	L.1 ------------------- - NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels. ------------------- Place channel in trip.	6 hours
M. Required Action and associated Completion Time of Condition L not met.	M.1 Be in MODE 3.	6 hours
	AND	
	M.2 Reduce pressurizer pressure to < 2000 psig.	12 hours
N. As required by Required Action A.1 and referenced by Table 3.3.2-1.	N.1 Declare associated Auxiliary Feedwater pump inoperable and enter applicable condition(s) of LCO 3.7.5, "Auxiliary Feedwater (AFW) System."	Immediately

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2	Perform COT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.2.3	- NOTE - Verification of relay setpoints not required. Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.4	- NOTE - Verification of relay setpoints not required. Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.5	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.6	Verify the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions are not bypassed when pressurizer pressure > 2000 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.7	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more Functions with two required channels inoperable.	D.1 Restore one channel to OPERABLE status.	7 days
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Enter the Condition referenced in Table 3.3.3-1 for the channel.	Immediately
F. As required by Required Action E.1 and referenced in Table 3.3.3-1.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	6 hours 12 hours
G. As required by Required Action E.1 and referenced in Table 3.3.3-1.	G.1 Initiate action to prepare and submit a special report.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -

SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.2	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

- NOTE -

When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 4 hours provided the second channel maintains LOP DG start capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.2	Perform CHANNEL CALIBRATION with Limiting Safety System Settings (LSSS)[a] for each 480 V bus as follows: a. Loss of voltage LSSS ≥ 372.0 V and ≤ 374.8 V with a time delay of ≥ 2.13 seconds and ≤ 2.62 seconds. b. Degraded voltage LSSS ≥ 420.0 V and ≤ 423.6 V with a time delay of ≥ 68.1 seconds and ≤ 125 seconds (@ 420 V) and ≥ 71.8 seconds and ≤ 125 seconds (@ 423.6 V).	In accordance with the Surveillance Frequency Control Program

(a)

A channel is OPERABLE when both of the following conditions are met:

1. The absolute difference between the as-found Trip Setpoint (TSP) and the previous as-left TSP is within the CHANNEL CALIBRATION Acceptance Criteria. The CHANNEL CALIBRATION Acceptance Criteria is defined as:

   |as-found TSP - previous as-left TSP| ≤ CHANNEL CALIBRATION uncertainty

   The CHANNEL CALIBRATION uncertainty shall not include the calibration tolerance.

2. The as-left TSP is within the established calibration tolerance band about the nominal TSP. The nominal TSP is the desired setting and shall not exceed the LSSS. The LSSS and the established calibration tolerance band are defined in accordance with the Ginna Instrument Setpoint Methodology. The channel is considered operable even if the as-left TSP is non-conservative with respect to the LSSS provided that the as-left TSP is within the established calibration tolerance band.

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.5-1 to determine which SRs apply for each Containment Ventilation Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.3	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.4	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.2	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.3	- NOTE - Verification of setpoint is not required. Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.4	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.5	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEMS (RCS)

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

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR.

- NOTE -

Pressurizer pressure limit does not apply during pressure transients due to:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

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	Verify pressurizer pressure is within limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.2	Verify RCS average temperature is within limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

	SURVEILLANCE	FREQUENCY
SR 3.4.1.3	- NOTE - Required to be performed within 7 days after ≥ 95% RTP. Verify RCS total flow rate is within the limit specified in the COLR.	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 Each RCS loop average temperature (T_{avg}) shall be ≥ 540°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or both RCS loops not within limit.	A.1 Be in MODE 2 with K_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 540°F.	Within 30 minutes prior to achieving criticality.
SR 3.4.2.2	----- - NOTE - Only required if any RCS loop T_{avg} < 547°F and the low T_{avg} alarm is either inoperable or not reset. ----- Verify RCS T_{avg} in each loop ≥ 540°F.	Once within 30 minutes and in accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODE 1 > 8.5% RTP

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

APPLICABILITY: MODE 1 > 8.5% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 1 $\leq$ 8.5% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Both RCS loops inoperable. OR No RCS loop in operation.	C.1 De-energize all CRDMs.	Immediately
	AND	
	C.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.	Immediately
	AND	
	C.3 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	Verify steam generator secondary side water levels are ≥ 16% for two RCS loops.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required RCP that is not in operation.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One RHR loop inoperable. AND Two RCS loops inoperable.	- NOTE - Required Action B.1 is not applicable if all RCS and RHR loops are inoperable and Condition C is entered. B.1 Be in MODE 5.	 24 hours
C. All RCS and RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. AND C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	Verify SG secondary side water level is $\geq$ 16% for each required RCS loop.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.4	-NOTE- Not required to performed until 12 yours after entering MODE 4. Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify SG secondary side water level is $\geq$ 16% in the required SG.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.4	Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.2	Verify correct breaker alignment and indicated power are available to the RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.3	Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Pressurizer water level not within limit.	A.1	Be in MODE 3 with reactor trip breakers open.	6 hours
		AND		
		A.2	Be in MODE 4.	12 hours
B.	Pressurizer heaters capacity not within limits.	B.1	Be in MODE 3.	6 hours
		AND		
		B.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is $\leq 87\%$.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify total capacity of the pressurizer heaters is ≥ 100 Kw.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.11.1	- NOTE - Not required to be performed with block valve closed per LCO 3.4.13. Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.12.1	- NOTE - Only required to be performed when complying with LCO 3.4.12.a. Verify no SI pump is capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	- NOTE - Only required to be performed when complying with LCO 3.4.12.b. Verify a maximum of one SI pump is capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	- NOTE - Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. Verify each ECCS accumulator motor operated isolation valve is closed.	Once within 12 hours and in accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	- NOTE - Only required to be performed when complying with LCO 3.4.12.b. Verify RCS vent $\geq$ 1.1 square inches open.	In accordance with the Surveillance Frequency Control Program for unlocked open vent valve(s) <u>AND</u> In accordance with the Surveillance Frequency Control Program for locked open vent valve(s)

SURVEILLANCE		FREQUENCY
SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.6	- NOTE - Required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR. Perform a COT on each required PORV, excluding actuation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.7	- NOTE - Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. Verify power is removed from each ECCS accumulator motor operated isolation valve operator.	Once within 12 hours and in accordance with the Surveillance Frequency Control Program
SR 3.4.12.8	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	- NOTE - 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. Verify leakage from each SI cold leg injection line and each RHR RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve

	SURVEILLANCE	FREQUENCY
SR 3.4.14.2	- NOTE - 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided.	
	Verify leakage from each SI hot leg injection line RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of containment atmosphere radioactivity monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform COT of containment atmosphere radioactivity monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment sump monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL CALIBRATION of containment atmosphere radioactivity monitors.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	- NOTE - Only required to be performed in MODE 1. Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 10 hours after a THERMAL POWER change of $\geq$ 15% RTP within a 1 hour period
SR 3.4.16.3	- NOTE – Only required to be performed in MODE 1. Determine $\bar{E}$ from a reactor coolant sample.	Once within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours. AND In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Two ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > 1600 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce pressurizer pressure to $\leq$ 1600 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator motor operated isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	Verify borated water volume in each accumulator is $\geq$ 1090 cubic feet (24%) and $\leq$ 1140 cubic feet (83%).	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 700 psig and ≤ 790 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2550 ppm and ≤ 3050 ppm.	In accordance with the Surveillance Frequency Control Program (by inleakage monitoring) <u>AND</u> In accordance with the Surveillance Frequency Control Program (by sample)
SR 3.5.1.5	Verify power is removed from each accumulator motor operated isolation valve operator when pressurizer pressure is > 1600 psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE				FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position.			In accordance with the Surveillance Frequency Control Program
	Number	Position	Function	
	825A	Open	RWST Suction to SI Pumps	
	825B	Open	RWST Suction to SI Pumps	
	826A	Closed	BAST Suction to SI Pumps	
	826B	Closed	BAST Suction to SI Pumps	
	826C	Closed	BAST Suction to SI Pumps	
	826D	Closed	BAST Suction to SI Pumps	
	851A	Open	Sump B to RHR Pumps	
	851B	Open	Sump B to RHR Pumps	
	856	Open	RWST Suction to RHR Pumps	
	878A	Closed	SI Injection to RCS Hot Leg	
	878B	Open	SI Injection to RCS Cold Leg	
	878C	Closed	SI Injection to RCS Hot Leg	
	878D	Open	SI Injection to RCS Cold Leg	
	896A	Open	RWST Suction to SI and Containment Spray	
	896B	Open	RWST Suction to SI and Containment Spray	
SR 3.5.2.2	-----------------NOTE------------------- Not required to be met for system vent flow paths opened under administrative control. --- 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 3.5.2.3	Verify each breaker or key switch, as applicable, for each valve listed in SR 3.5.2.1, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Verify, by visual inspection, each RHR containment sump suction inlet is not restricted by debris and the containment sump screen shows no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.8	Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST water volume not within limits.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water volume is ≥ 300,000 gallons (88%).	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST boron concentration is ≥ 2750 ppm and ≤ 3050 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 Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in each 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 mini-purge valve is closed, except when the penetration flowpath(s) are permitted to be open under administrative control.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	- NOTE - 1. Isolation boundaries in high radiation areas may be verified by use of administrative controls. 2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal. Verify each containment isolation boundary that is located outside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation accident function except for containment isolation boundaries that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	- NOTE - 1. Isolation boundaries in high radiation areas may be verified by use of administrative means. 2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal. Verify each containment isolation boundary that is located inside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation accident function, except for containment isolation boundaries 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 containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Perform required leakage rate testing of containment mini-purge valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Program.

SURVEILLANCE		FREQUENCY
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the required position actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ 1.0 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

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

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 125°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CS trains inoperable. OR Three or more CRFC units inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B.	In accordance with applicable SRs.
SR 3.6.6.2	-----NOTE----- Not required to be met for system vent flow paths opened under administrative control. ---------- Verify each CS 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 3.6.6.3	Verify each NaOH 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 3.6.6.4	Operate each CRFC unit for $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5	Verify cooling water flow through each CRFC unit.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify each CS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.7	Verify NaOH System solution volume is $\geq$ 3000 gal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.8	Verify NaOH System tank NaOH solution concentration is $\geq$ 30% and $\leq$ 35% by weight.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.6.6.9	Perform required CRFC unit testing in accordance with the VFTP.	In accordance with the VFTP
SR 3.6.6.10	Verify each automatic CS valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.11	Verify each CS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.12	Verify each CRFC unit starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.13	Verify each automatic NaOH System valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.14	Verify spray additive flow through each eductor path.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.15	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage
SR 3.6.6.16	Verify CS locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify closure time of each MSIV is ≤ 5 seconds under no flow and no load conditions.	In accordance with the Inservice Testing Program
SR 3.7.2.2	Verify each main steam non-return check valve can close.	In accordance with the Inservice Testing Program
SR 3.7.2.3	Verify each MSIV can close on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Two ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System average temperature (T_{avg}) $\geq$ 500°F.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One ARV line inoperable.	A.1	Restore ARV line to OPERABLE status.	7 days
B.	Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3 with T_{avg} < 500°F.	8 hours
C.	Two ARV lines inoperable.	C.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Perform a complete cycle of each ARV.	In accordance with the Surveillance Frequency Control Program
SR 3.7.4.2	Verify one complete cycle of each ARV block valve.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW and SAFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the 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 - Required to be met prior to entering MODE 1 for the TDAFW pump. Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3	Verify the developed head of each SAFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.4	Perform a complete cycle of each AFW and SAFW motor operated suction valve from the Service Water System, each AFW and SAFW discharge motor operated isolation valve, and each SAFW cross-tie motor operated valve.	In accordance with the Inservice Testing Program
SR 3.7.5.5	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.6	- NOTE - Required to be met prior to entering MODE 1 for the TDAFW pump. Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.7	Verify each SAFW train can be actuated and controlled from the control room.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LCO 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST water volume not within limit.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours
	AND	
	A.2 Restore CST water volume to within limit.	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.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST water volume is $\geq$ 24,350 gal.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
	D.2	Be in MODE 3.	6 hours
	AND		
	D.3	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	- NOTE - Isolation of CCW flow to individual components does not render the CCW loop header inoperable.	
	Verify each CCW manual and power operated valve in the CCW train and heat exchanger flow path and loop header 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	Perform a complete cycle of each motor operated isolation valve to the residual heat removal heat exchangers.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify screenhouse bay water level and temperature are within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	- NOTE - Isolation of SW flow to individual components does not render the SW loop header inoperable. Verify each SW manual, power operated, and automatic valve in the SW flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify all SW loop header cross-tie valves are locked in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Verify each SW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.5	Verify each SW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

CONDITION		REQUIRED ACTION		COMPLETION TIME
D.	Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1	Place OPERABLE CREATS train in emergency mode.	Immediately
		<u>OR</u>		
		D.2	Suspend movement of irradiated fuel assemblies.	Immediately
E.	Two CREATS trains inoperable during movement of irradiated fuel assemblies. <u>OR</u> One or more CREATS trains inoperable due to an inoperable CRE boundary during movement of irradiated fuel assemblies.	E.1	Suspend movement of irradiated fuel assemblies.	Immediately
F.	Two CREATS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREATS filtration train $\geq$ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.2	Perform required CREATS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify each CREATS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.10 Auxiliary Building Ventilation System (ABVS)

LCO 3.7.10 The ABVS shall be OPERABLE and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the Auxiliary Building when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ABVS inoperable.	A.1 - NOTE - LCO 3.0.3 is not applicable. Suspend movement of irradiated fuel assemblies in the Auxiliary Building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify ABVS is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Verify ABVS maintains a negative pressure with respect to the outside environment at the Auxiliary Building operating floor level.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.3	Perform required Spent Fuel Pool Charcoal Adsorber System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool (SFP) Water Level

LCO 3.7.11 The SFP 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 SFP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify the SFP water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.12 The SFP boron concentration shall be ≥ 2300 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the SFP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify the SFP pool boron concentration is within limit.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Secondary Specific Activity

LCO 3.7.14 The specific activity of the secondary coolant shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the specific activity of the secondary coolant is $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program

CONDITION		REQUIRED ACTION		COMPLETION TIME
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
		AND		
		D.2	Be in MODE 5.	36 hours
E.	Two DGs inoperable.	E.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 the offsite circuit to each of the 480 V safeguards buses.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	- NOTE - 1. Performance of SR 3.8.1.9 satisfies this SR. 2. 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 achieves rated voltage and frequency.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	- NOTE - 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.9. Verify each DG is synchronized and loaded and operates for ≥ 60 minutes and < 120 minutes at a load ≥ 2025 kW and < 2250 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify the fuel oil level in each day tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Verify the DG fuel oil transfer system operates to transfer fuel oil from each storage tank to the associated day tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.6	Verify transfer of AC power sources from the 50/50 mode to the 100/0 mode and 0/100 mode.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	- NOTE - 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. Verify each DG does not trip during and following a load rejection of ≥ 295 kW.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	- NOTE - 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. Verify each DG automatic trips are bypassed on an actual or simulated safety injection (SI) signal except: a. Engine overspeed; b. Low lube oil pressure; and c. Start failure (overcrank) relay.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.9	- NOTE - 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal: a. De-energization of 480 V safeguards buses; b. Load shedding from 480 V safeguards buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads, 2. energizes auto-connected emergency loads through the load sequencer, and 3. supplies permanently and auto-connected emergency loads for $\geq$ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains $\geq$ 5000 gal of diesel fuel oil for each required DG.	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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 129 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	- NOTE - 1. SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 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.	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. Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test.	In accordance with the Surveillance Frequency Control Program <u>AND</u> 12 months when battery shows degradation, or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for Train A and Train B batteries shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DC electrical power sources are required to be OPERABLE by LCO 3.8.5, "DC Sources - MODES 5 and 6."

ACTIONS

- NOTE -

Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within limits.	A.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.6.1	Verify electrolyte level of each connected battery cell is above the top of the plates and not overflowing.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.2	Verify the float voltage of each connected battery cell is > 2.07 V.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.3	Verify specific gravity of the designated pilot cell in each battery is ≥ 1.195.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.4	Verify average electrolyte temperature of the designated pilot cell in each battery is ≥ 55°F.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.6.5	Verify average electrolyte temperature of every fifth cell of battery is ≥ 55°F.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.6	Verify specific gravity of each connected battery cell is: a. Not more than 0.020 below average of all connected cells, and b. Average of all connected cells is ≥ 1.195.	In accordance with the Surveillance Frequency Control Program

CONDITION		REQUIRED ACTION		COMPLETION TIME
D.	Two or more required instrument bus sources inoperable.	D.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct static switch alignment to Instrument Bus A and C.	In accordance with the Surveillance Frequency Control Program
SR 3.8.7.2	Verify correct Class 1E CVT alignment to Instrument Bus B.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.8.1	Verify correct static switch alignment to required AC instrument bus(es).	In accordance with the Surveillance Frequency Control Program
SR 3.8.8.2	Verify correct Class 1E CVT alignment to the required AC instrument bus.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required electrical power trains.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
		AND	
	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 electrical power distribution train(s) to OPERABLE status.	Immediately
		AND	
	A.2.5	Declare associated required residual heat removal loop(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required electrical power distribution trains.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
	C.2	Suspend positive reactivity additions.	Immediately
	AND		
	C.3	Perform SR 3.9.1.1	4 hours AND Once per 12 hours thereafter

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.2	Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.3	Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

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 and determining CRE unfiltered inleakage as required by paragraph c.

5.5.17 Surveillance Frequency Control Program

This program provides controls for the 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 the Surveillance Requirements for which the Frequency is controlled by the program.

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 122, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

RHyman for,

Travis L. Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: June 28, 2016.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 122

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

R. E. GINNA NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC.

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

1.0 INTRODUCTION

By application dated June 4, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15166A075), as supplemented by letters dated February 3, 2016, March 29, 2016, and June 16, 2016 (ADAMS Accession Nos. ML16034A139, ML16089A425, and ML16168A261 respectively), Exelon Generation Company, LLC (the licensee) requested changes to the Technical Specifications (TSs) for Ginna Nuclear Power Plant (Ginna) reference [1], which is contained in Appendix A of Renewed Facility Operating License DPR 18. The licensee requested to revise the Ginna TS by relocating specific surveillance requirements (SRs) frequencies to a licensee-controlled program. The licensee requested to revise the TSs to require that changes to such surveillance frequencies will be made 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" (ADAMS Accession No. ML071360456). The requested change is the adoption of Revision 3 of the U.S. Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-425, "Relocate Surveillance Frequencies to Licensee Control-RITSTF [Risk-Informed TSTF] Initiative 5b" (ADAMS Accession No. ML090850642). The *Federal Register* (FR) notice published on July 6, 2009 (74 FR 31996), announced the availability of Revision 3 of TSTF-425.

The NRC sent a letter with requests for additional information (RAIs) to the licensee on January 7, 2016 (ADAMS Accession No. ML15344A353). The supplemental response letters dated February 3, 2016, and March 29, 2016, provided clarifying information that did not expand the scope of the application and did not change the staff's original proposed no significant hazards consideration determination as published in the FR on October 13, 2015 (80 FR 61482).

Enclosure

2.0 REGULATORY EVALUATION

2.1 Description of the Proposed Changes

The licensee proposed to modify the Ginna TSs by relocating specific surveillance frequencies to a licensee-controlled program (i.e., the Surveillance Frequency Control Program (SFCP)) in accordance with Revision 1 of NEI 04-10. The licensee stated that the proposed change is consistent with the adoption of NRC-approved Revision 3 of TSTF-425. When implemented, Revision 3 of TSTF-425 relocates most periodic frequencies of TS surveillances to the SFCP and provides requirements for the new program in the Administrative Controls sections of the TSs. All surveillance frequencies can be relocated except the following:

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

The licensee proposed to add the SFCP to Subsection 5.5, "Programs and Manuals", in the Administrative Controls section of the TSs. The SFCP describes the requirements for the program to control changes to the relocated surveillance frequencies. The TS Bases for each affected surveillance would be revised to state that the frequency is controlled under the SFCP. In order to incorporate the SFCP in the Administrative Controls section of the TSs, Ginna proposed a change to include a specific reference to Revision 1 of NEI 04-10, as the basis for making any changes to the surveillance frequencies once they are relocated out of the TSs.

In a letter dated September 19, 2007 (ADAMS Accession No. ML072570267), the NRC staff approved Revision 1 of Topical Report NEI 04-10, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, Risk-informed Technical Specification Initiative 5b, "Risk-Informed Method for Control of Surveillance Frequencies." Revision 1 of NEI 04-10 was approved as acceptable for referencing in licensing actions to the extent specified, under the limitations delineated, and the safety evaluation (SE) provided the basis for NRC's acceptance.

The licensee's changes and deviations from TSTF-425 are discussed in Section 3.3 of this SE.

2.2 Applicable Commission Policy Statements

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

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed.
>
> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "* * * probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made * * * about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety."
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

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

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

of traditional regulation by considering risk in a more coherent and complete manner...

Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth 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 Section 50.36 of 10 CFR, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs);

(3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. These categories will remain in the Ginna TSs.

Paragraph 50.36(c)(3) in 10 CFR states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The FR notice published on July 6, 2009 (74 FR 31996), which announced the availability of Revision 3 of TSTF-425, 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 Revision 1 of NEI 04-10 including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), and Criterion XVI, "Corrective Action", in Appendix B of 10 CFR Part 50 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 Revision 1 of NEI 04-10, the licensee will be required to monitor the performance of SSCs which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

2.4 Applicable NRC Regulatory Guides and Review Plans

Revision 2 of Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML100910006), 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.

Revision 1 of RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML100910008), describes an acceptable risk-informed approach specifically for assessing proposed TS changes.

Revision 2 of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors (LWRs).

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 19.2 in

Chapter 19, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (ADAMS Accession No. ML071700658). Guidance on evaluating PRA technical adequacy is provided in Revision 3 of Section 19.1 of the SRP, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests [LARs] After Initial Fuel Load" (ADAMS Accession No. ML12193A107). More specific guidance related to risk-informed TS (RITS) changes is provided in Revision 1 of Section 16.1 of the SRP, "Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML070380228), which includes changes to surveillance test intervals (STIs) (i.e., surveillance frequencies) as part of risk-informed decision making. Section 19.2 of the SRP references the same criteria as Revision 1 of RG 1.177 and Revision 2 of RG 1.174, 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 explicitly relates 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 core damage frequency (CDF) or risk, the increase(s) 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.

3.0 TECHNICAL EVALUATION

The licensee's adoption of Revision 3 of TSTF-425 provides for administrative relocation of applicable surveillance frequencies 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 Revision 1 of NEI 04-10 for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes described in Revision 3 of TSTF-425 included documentation regarding the PRA technical adequacy consistent with Revision 2 of RG 1.200. Revision 1 of NEI 04-10 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 Revision 2 of RG 1.174 and Revision 1 of RG 1.177, in support of changes to STIs.

3.1 Review Methodology

Revision 1 of RG 1.177 identifies five key safety principles required for risk-informed changes to TSs. Each of these principles is addressed by Revision 1 of NEI 04-10.

3.1.1 The Proposed Change Meets Current Regulations

The licensee is required by paragraph 50.36(c)(3) of 10 CFR 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 Revision 1 of NEI 04-10 provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

This change will follow paragraph 50.36(c)(3) in 10 CFR by keeping the SRs in TSs, but will relocate the related surveillance frequencies to licensee-controlled documents. Examples of surveillances being relocated are those performed in accordance with the In-Service Testing Program and the Primary Containment Leakage Rate Testing Program. Thus, this proposed change complies with paragraph 50.36(c)(3) in 10 CFR by retaining the requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

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

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

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

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

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

- Independence of barriers is not degraded.

- Defenses against human errors are preserved.

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

The changes to the Administrative Controls section of the TSs will require the application of Revision 1 of NEI 04-10 for any changes to surveillance frequencies within the SFCP. The large early release frequency (LERF) and CDF metrics in Revision 1 of NEI 04-10 are used to evaluate the impact of proposed changes to surveillance frequencies. The guidance in Revision 2 of RG 1.174 and Revision 1 of RG 1.177 for changes to CDF and LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and 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 staff concludes that both the quantitative risk analysis and the qualitative considerations assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle in Revision 1 of RG 1.177.

3.1.3 <u>The Proposed Change Maintains Sufficient Safety Margins</u>

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

The design, operation, testing methods, and acceptance criteria for SSCs specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant's licensing bases, including the Updated Final Safety Analysis Report and TS Bases, because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant's licensing basis. On this basis, the staff concludes that safety margins are maintained by the proposed methodology, and the third key safety principle of Revision 1 of RG 1.177 is satisfied.

3.1.4 <u>When Proposed Changes Result in an Increase in CDF or Risk, the Increases Should Be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement</u>

The framework for evaluating the risk impact of proposed changes to surveillance frequencies which requires identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations is provided in Revision 1 of RG 1.177. The SFCP will require the application of Revision 1 of NEI 04-10 to changes in the Administrative Controls section of the TSs. The intention of the guidance in Revision 1 of RG 1.177 is satisfied by Revision 1 of NEI 04-10 for evaluation of the change in risk, and for assuring that such changes

are small by providing the technical methodology to support RITSs 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.

Regulatory guidance for assessing the technical adequacy of a PRA is provided by RG 1.200. The use of (1) American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (i.e., the PRA Standard), (2) NEI 00-02, "PRA Peer Review Process Guidance" (ADAMS Accession Nos. ML061510619 and ML063390593), and (3) NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (ADAMS Accession No. ML083430462) is endorsed by Revision 2 of RG 1.200 with clarifications and qualifications.

The licensee has performed an assessment of the PRA models used to support the SFCP using the guidance in Revision 2 of RG 1.200. Plant-specific data and models are used in the assessment to assure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs. Capability Category (CC) II of the NRC-endorsed 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 assessed further to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies where appropriate, in accordance with Revision 1 of NEI 04-10.

In Attachment 2 of the LAR the licensee states:

> The Ginna PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the Ginna PRA is suitable for use in risk-informed processes, such as that proposed for the implementation of a [SFCP].

Exelon Generation based the above conclusion that the Ginna PRA was technically adequate on the following three factors:

(1) The 2009 internal events and the 2012 Fire PRA (FPRA) model peer reviews.

(2) The satisfactory disposition of findings and comments from both peer reviews and the commitment to assess the impact of any remaining gaps on the STI application, and if necessary, perform sensitivity analysis on this impact.

(3) The adequacy of the Exelon PRA maintenance and update process and the commitment to review and assess any open items in the Updating Requirement Evaluation (URE) database for impact on each STI evaluation and to perform additional sensitivity studies, as needed.

The 2009 peer review was the latest and most applicable review of the internal events PRA. This peer review led to 24 Findings and Observations (F&Os), 11 of which remain open. The LAR addresses the importance of these open F&Os to the TSTF-425 application.

The LAR submittal includes discussion of the resolution of the peer review F&Os that are applicable to the parts of the PRA required for the application, which is required by Section 4.2 in Revision 2 of RG 1.200. This should take the following forms: (i) a discussion of how the PRA model has been changed and (ii) a justification, in the form of a sensitivity study, which demonstrates that the accident sequences or significant contributors to the application decision were not adversely impacted (i.e., they remained the same) by the particular issue.
This requirement is acknowledged in Section 1, "Overview," in Attachment 2 of the LAR by their statement:

4. Demonstrate the Technical Adequacy of the PRA.

   Document peer review [F&Os] that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.

An assessment of the 11 open findings from the 2009 peer review were provided in Ginna's LAR, but was unclear on how previously closed peer review findings were resolved relative to regulatory applications for RITS surveillance frequencies. Additionally, there was no gap assessment provided against Revision 2, versus Revision 1, of RG 1.200 against which the 2009 peer review was conducted. To clarify these items the NRC asked for additional information in the 17 RAIs sent to the licensee as reference [2]. Ginna responded to these RAIs in references [3] and [4].

The responses to the 17 RAIs were evaluated as discussed below. All original responses for RAIs were found to be acceptable, except 2.c, 2.d.iv, 3 and 15. Supplemental requests to clarify or enhance these four responses were satisfactorily answered and approved as acceptable. Each individual RAI and completed response is discussed below. References 2 and 3 provide the detailed documentation of the RAIs transmitted and the responses received from the licensee.

RAI 1

Assessment of the technical adequacy of the Ginna Internal Events PRA is based primarily on the 2009 peer review in Attachment 2 of the LAR. As required by Revision 2 of RG 1.200, document all the individual findings and selected suggestions, i.e., those suggestions for which the reference supporting requirements changed between the 2007 version of the ASME/ANS PRA Standard, as clarified by Revision 1 to RG 1.200, and the 2009 version of the Standard, as clarified and qualified by Revision 2 of RG 1.200, resulting from the 2009 internal events peer review, and their disposition, whether or not they have been closed (unless closed via a subsequent peer review, full or focused-scope). Include discussion as to whether the disposition applies to changes in risk as well as the base-line risk, since the peer review is against the latter, but the application involves the former as well.

Exelon Response to RAI 1

Table 2-1 from the TSTF-425 LAR reference [1] has been updated to disposition the findings with respect to their current status and to note the potential impact on changes in risk as well as base-line risk. Changes compared to the original Table 2-1 are shown in italics. Table 2-1 provided in this submittal supersedes the original Table 2-1 submitted on June 4, 2015.

Of the 34 suggestions from the Ginna peer review, excluding formatting and very minor editorial changes, only 9 of the suggestions were associated with changes to the reference supporting requirements which changed between the 2007 version of the [ASME/ANS] PRA Standard, as clarified by Revision 1 of RG 1.200, and the 2009 version of the Standard, as clarified and qualified by Revision 2 of RG 1.200. The current disposition of the applicable suggestions is provided in Table 2-2.

NRC Evaluation of Exelon Response to RAI 1

Based on the information in the updated table 2-1 and table 2-2 and related discussion provided by the licensee, the response to RAI 1 is acceptable for use in the TSTF-425 program.

RAI 2

The LAR indicates that a FPRA, associated with transition to National Fire Protection Association (NFPA)-805, was performed and peer reviewed in August 2012. However, the F&Os identified from the NFPA-805 fire peer review were not provided for consideration in the LAR associated with RITS-5b changes to TS surveillance frequencies. The LAR states:

> The 2012 [FPRA] peer review for the PRA ASME model update identified 183 [s]upporting [r]equirements to be reviewed for the Ginna PRA. Of these 2 were not met, 2 met [CC] 1, 8 partially met CC 2, 17 met CC 2, 13 partially met CC 3, 7 met CC 3, and 118 fully met all capability requirements and 16 were not applicable. There were 19 findings and 22 suggestions issued to address potential gaps to compliance with the PRA standard. There were 3 Best Practices. All of the findings from the [FPRA] peer review have since been closed. As the results of this peer review have already been communicated to the NRC as part of the NFPA-805 submittal and subsequent [RAIs], these will not be catalogued in this document.

Previous responses, described above, provided in the NFPA-805 submittals are associated with assessing the PRA technical adequacy to address fire-related hazards. To the extent that there were deficiencies in the FPRA models associated with SSCs for which changes to TS surveillance frequencies are being sought, there is no equivalent clarification of how the FPRA related F&Os will not have an impact on the Revision 3 of TSTF-425. It is the NRC's position that FPRA related F&Os must be considered when evaluating TS surveillance frequency changes. The NRC asked for additional information on this in a formal RAI to provide the following:

a. An assessment of how the 2012 fire peer review F&Os have been resolved to assure PRA Technical Adequacy with respect to TSTF-425, not NFPA-805. Include discussion

as to whether the disposition applies to changes in risk, as well as the base-line risk, since the peer review is against the latter, but the application involves the former as well.

b. For those FPRA related F&Os, which are dispositioned as not having an impact on Revision 3 of TSTF-425, provide the technical basis for this determination.

c. Discussion of how the licensee plans to incorporate updates to FPRA state-of-the-art enacted since the 2012 peer review, including but not limited to updated fire ignition frequencies and non-suppression probabilities (as per NUREG–2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database") and updated spurious operation occurrence probabilities and probabilities for duration exceedance (as per NUREG/CR–7150, Volume 2, "Joint Assessment of Cable Damage and Quantification of Effects from Fire.")

   This was supplemented by the following:

   In the response, NUREG-2178, "Refining and Characterizing Heat Release Rates from Electrical Enclosures during Fire (RACHELLE-FIRE)," was among three source documents cited as affecting fire frequency development, specifically by "reduc[ing] the heat release rates, [thereby] reducing the electrical cabinet high risk scenario frequencies."

   > Was NUREG-2178 <u>only used</u> to re-partition enclosures (e.g., self-extinguishing fires vs. fires that propagate)? If so, was this the means by which it was used to alter fire <u>ignition</u> frequencies?
   >
   > Additionally, it is stated that the cited effects <u>"should"</u> result in a net reduction of the higher risk scenarios and, as a result, "the existing NFPA-805 model is conservative with regard to TSTF-425 delta risk calculations."
   >
   > Is this conclusion of conservatism definitive, or an expectation, given the use of "should" vs. "shall?"

d. Consistent with the requirements of Table A-4 in Revision 2 of RG 1.200, clarify how the FPRA addresses the following requirements with regard to differential risk evaluations related to Revision 3 of TSTF-425:

   i. In SR FSS-A4, Revision 2 of RG 1.200 changes "one of more" to "sufficient."

   ii. In Fire PRA F&O FSS-F1-01, Revision 2 of RG 1.200 changes SR FSS-F1 from "one or more fire scenarios that could" to "a sufficient number of fire scenarios to characterize."

   iii. In Fire PRA F&O FSS-G5-01: Is potential failure of the wall water spray system to provide structural integrity of the boundary addressed? This includes the probability that the system does not perform its function such that the boundary could be breached and result in a multi-compartment fire scenario (e.g., the assumption of perfect reliability versus high reliability, is non-conservative).

iv. In Fire PRA F&O SF-A1-02, provide a disposition that addresses the item of concern, namely failure of the analysis to fully assess the potential impact of a seismically induced failure (rupture or spurious operation of fire protection features on the post-earthquake response.

This was supplemented by the following.

> The response does not appear to specifically cite the item of concern in Fire PRA F&O SF-A1-02, namely the failure to address the impact of seismically-induced rupture or spurious operation of fire protection features on post-earthquake plant responses. Presumably, such responses would include operator actions that could be affected due to inadvertent actuation of suppression systems or other extinguishing agents in areas where not needed to fight a fire (e.g., impeding access). Confirm that such operator actions would not be affected.

v. The disposition of SR FSS-G5 partly justifies reclassifying the F&O as CC II based on the disposition cited for F&O FSS-G5-01 discussed previously. The concern discussed previously needs to be resolved in order for the CC II assignment to be fully justified.

Exelon Response to RAI 2.a and 2.b

Although the [p]eer [r]eview was focused on the baseline risk, the [NFPA]-805 submittal required both acceptable baseline risks as well as an acceptable delta risks. Knowing this requirement, the closure of the [FPRA] findings focused on addressing the finding versus using a conservative argument that could mask delta risk calculations. As such, the NFPA-805 dispositions would also support TSTF-425.

All of the findings in Table V-1 of the NFPA-805 LAR submittal were reviewed again to specifically assess if the finding closure could introduce conservatisms that could significantly affect the TSTF-425 delta risk calculations. There are some inherent conservatisms associated with the NFPA-805 methods, but only conservatisms beyond those are identified. This is appropriate given the NFPA-805 methods are deemed acceptable.

Of the findings listed in Table V-1, there are only two findings listed with conservative closure practices. These two findings are dispositioned for TSTF-425 acceptability:

- FSS-A3-01 - bounding cable routes used - Although bounding routes were used for some conduits, bounding routes that significantly affected risk calculations were further walked down to refine the routing. Due to the limited role the remaining bounding routes play in the analysis, this will not significantly affect delta risk calculations.

- FSS-A6-01 - conservative Main Control Room (MCR) frequency development - This approach does make the control room risk more important than a traditional NUREG/CR 6850 Appendix L approach. But, this is only a frequency issue and does not mask any equipment impacts that would be evaluated in a TSTF-425 delta risk calculation. This MCR modeling just increases the calculated delta risks.

The remaining finding closures were largely documentation improvements or findings that were directly resolved without introducing any conservatisms that could potentially mask a TSTF-425 delta risk.

NRC Evaluation of Exelon Response to RAIs 2.a and 2.b

The NRC finds the licensee's response to RAIs 2.a and 2.b acceptable for the program because the licensee reviewed the FPRA F&Os and refined its analysis for use in the TSTF-425 program.

Exelon Response to RAI 2.c

The [Volume 2 of] NUREG/CR-7150 information was already included in the NFPA-805 analysis. The next revision of the fire model will incorporate [Appendix L of] NUREG/CR 6850, NUREG-2169, and NUREG-2178. All three of these changes affect fire scenario frequency development. NUREG-2169 will cause an increase in the Main Control Board (MCB) and electrical cabinet frequencies. NUREG-2178 will reduce the Heat Release Rates (HRRs) of most of the electrical cabinets on the site. Reducing HRRs of an electrical cabinet will cause a corresponding reduction in the associated severity factor (i.e., damage is limited to the ignition source). This causes a consequent increase in the frequencies of only the ignition source being affected (i.e., the low risk significant scenario). That frequency increase is then removed from the associated scenario where damage occurs beyond the ignition source (i.e. the higher risk scenario). As a result, the frequency of ignition-source-damage-only increases while the frequency of damage-beyond-the-ignition source decreases. This causes a net reduction in risk due to electrical cabinet fires. Appendix L [of NUREG/CR 6850] will lower the MCB frequencies. These three changes in aggregate should result in a net reduction in risk. As a result, the existing NFPA-805 model is believed to be conservative with regard to TSTF-425 delta risk calculations.

However, as these changes are all frequency related, no masking issues are introduced or removed by these updates. Issues can be masked when a conservative assumption is made that prevents changes in the risk from being seen by the risk model. An example would be if the main feed water system were assumed to fail on every trip (this is an example only and is not the case at Ginna). This would mask any surveillance frequency impacts that could affect the failure likelihood of the Main Feedwater System. A risk increase would exist but the [Conditional Core Damage Probability] CCDP and [Conditional Large Early Release Probability] CLERP would not result in an increase. As the NUREG updates (6850, 2169, and 2178) only affect fire scenario frequencies and not the CCDP or CLERP for a given scenario, no masking is introduced.

NRC Evaluation of Exelon Response to RAI 2.c

Based on the information described above, the NRC staff finds that the licensee has and will be able to identify any updates to the FPRA model based on state of the art information.

Exelon Response to RAI 2.d.i

Sufficient targets have been identified for the ignition sources in the unscreened Physical Analysis Units (PAUs) such that the credible range of system and function impacts has been

represented. A typical range of impacts is the ignition source, the ignition source plus a set of raceways and adjacent equipment, and a full compartment burn. If the full compartment burn contribution is too large, then intermediate scenarios are developed if the key target raceways are fairly well removed from the ignition source.

NRC Evaluation of Exelon Response to RAI 2.d.i

Based on the licensee's description of a typical range of impacts in the PAUs, the NRC staff finds that the licensee can apply this supporting requirement to the TSTF-425 program.

Exelon Response to RAI 2.d.ii

A sufficient number of fire scenarios has been developed to characterize the damage leading to collapse of the exposed structural steel for each identified scenario. All of the ignition sources in non-full-compartment-burn PAUs that have a high enough heat release rate to damage exposed structural steel are included in the evaluation. This primarily includes oil fire scenarios.

NRC Evaluation of Exelon Response to RAI 2.d.ii

Based on the licensee's description of the fire scenarios considering sufficiency, the NRC staff finds that the licensee can apply this supporting requirement to the TSTF-425 program.

Exelon Response to RAI 2.d.iii and 2.d.v

As discussed in [Table V-1 of] the NFPA-805 [LAR], the water spray system is not credited as a boundary under NFPA-805 and is not allowed per the standard. However, this water spray system is a design requirement for Ginna. The NFPA-805 analysis uses the standard NUREG/CR-6850 approach of plant partitioning PAUs. There is a concrete wall between the turbine building and the control room which is an adequate barrier per NUREG/CR 6850. The spray system was installed from a design perspective for defense-in-depth given all combustibles in the turbine building is engaged in a fire. Although purely a design issue, it was requested by the peer review team that this additional information be provided. The suppression system is credited in the multi-compartment analysis with the appropriate reliability and availability factors considered.

NRC Evaluation of Exelon Response to RAI 2.d.iii and 2.d.v

The NRC staff finds this information acceptable for the TSTF-425 program because the licensee clarified how the water spray system was being applied to the PRA and the potential non-conservative reliability factor.

Exelon Response to RAI 2.d.iv

As discussed in [Table V-1 of] the NFPA-805 [LAR], all of the areas in the global analysis boundary were assessed and dispositioned as not having a significant seismic impact that is not already bounded by existing fire scenarios. The original F&O was a scoping issue only. The first part of the finding in Table V-1 of the NFPA-805 submittal report states: "The licensee sufficiently addressed (assessed) the potential impact of a seismically induced failure or

spurious operation of fire protection features on the performance of post-earthquake fire protection equipment and fire brigade." This finding was based on the fact that during the initial assessment the high radiation areas were not walked down. Although not walked down, these areas were assessed to ensure that the post-earthquake mitigation strategy for fire mitigation is acceptable. This review was documented in the updated version of the Seismic Fire Notebook. The review was the closure basis for the original finding as documented in [Table V-1 of the] NFPA-805 [LAR].

In regards to operator actions, Ginna has both FLEX related and major incident related procedures that direct the use of alternative means of fire suppression in the event the installed suppression systems are unavailable.

NRC Evaluation of Exelon Responses for RAI 2 and its subparts

Based on the discussion above, the NRC staff finds that the licensee can apply this supporting requirement to the TSTF-425 program.

RAI 3

Clarification to IE-C12 regarding resolution of F&O IE-C10-01 [SR IE-C12 in the current version of the ASME/ANS Risk Standard] is added by Revision 2 of RG 1.200, including its accompanying Note. Since this peer review finding was against Revision 1 of RG 1.200, explain whether there is any change to the disposition or impact on TSTF-425 as a result of the Revision 2 update. If none, justify why not.

This was supplemented by the following:

Confirm that the gap assessment for the current PRA model of record against Revision 2 of RG 1.200 concludes that F&O IE-C10-01 (related to IE-C12 in the 2009 version of the ASME/ANS PAA Standard) has been resolved.

Exelon Response to RAI 3

The Ginna Full Power Internal Events (FPIE) PRA has detailed modeling for several initiating events, with logic built into Support System Initiating Event (SSIE) fault trees. In several instances, events such as loss of [c]omponent [c]ooling [w]ater could have been modeled with a single initiating event (IE) rate. These annualized IE rates are obtained from the NRC's 2012 update to NUREG/CR-6928 values, which include industry data through 2010. SR IE-C12 from the ASME PRA standard states to compare these generic industry values with the equivalent quantified gate in the PRA. [Revision 2 of RG 1.200] includes guidance to "COMPARE results and EXPLAIN differences in the initiating event analysis with generic data sources to provide a reasonable check of the results." During the 2015 FPIE PRA Update, the IE Notebook was updated with a comparison of the IE values provided in Section 4.4.4.

In most cases, quantified Ginna SSIE values were comparable to generic data. The main differences were with electrical bus failures. Several SSIE fault trees have operator action recoveries. These recoveries, with the addition of more detailed modeling, caused the lower event frequencies in some initiating events. In some cases with electrical bus SSIE fault trees, logic could be simplified by just using an IE with the generic event rate. Beyond the

electrical bus initiating events, no significant differences were found and this should have little impact on TSTF-425 analysis. The difference in the loss of bus initiating events is captured as a[n] [URE] which will be reviewed for applicability for each STI change evaluation as required by Exelon procedural guidance.

Ginna PRA F&O IE-C10-01 (related to IE-C12 in the 2009 version of the ASME/ANS PRA Standard) has been resolved, and is now has a status of "complete."

NRC Evaluation of Exelon Response to RAI 3

The licensee's response addressed the gap between versions of the standard as clarified by the corresponding RG 1.200 revision. Therefore, the licensee's disposition is acceptable for use in the TSTF-425 program.

<u>RAI 4</u>

The current PRA model was assessed to only be CC I, whereas expectations are that all supporting requirements be met at the CC II level (or justification be provided for the adequacy of CC I for the specific application) regarding resolution of SC-A12-01 [Also SR SC-A-2 in the current version of the ASME/ANS Risk Standard], which remains unresolved. The position that the supporting requirement is conservative and that differential risk evaluations for the TS surveillance frequency changes will thus also be conservative is presented in the LAR. Verify by example, or analysis, that this presumed conservatism is such that it ensures the differential risk for the application is also conservative, (i.e., the risk estimated for the before versus after condition is not overestimated such that subtracting it from the after value could underestimate the risk increase).

Exelon Response to RAI 4

For most of the end-state success criteria cases using the thermal-hydraulic analysis software, core uncovery alone was used as a surrogate for core damage. A benchmarking analysis includes cases of core uncovery as well as core damage. In those cases, the difference between core uncovery and core damage was fairly short (e.g. a few minutes). For cases that lead to core uncovery, core heat removal is clearly lost or clearly maintained except for larger loss-of-coolant accidents [LOCAs]. In the case of Large [LOCAs], it is identified that core uncovery can initially occur, but the core can be quickly recovered with the accumulators and residual heat removal (RHR) pumps providing makeup, mitigating a core damage event. A differential risk calculation could be impacted if: (1) if a valid system recovery is not credited for re-covering the core prior to core damage; or (2) the time between core uncovery and core damage is significant where a previously unidentified system recovery could take place that is not credited in the model due to timing restraints. Although no other cases are identified where core uncovery does not lead to core damage in short order, this issue is captured as an [URE] which will be reviewed for applicability for each [STI] change evaluation as required by Exelon procedural guidance.

NRC Evaluation of Exelon Response to RAI 4

The original peer review F&O was focused on use of simplifying conservatisms in core damage success criteria. The licensee's response clarified their approach for the relationship between

accident success criteria, time available for recovery actions, systems credited for recovery and is thus acceptable for use in the TSTF-425 program.

RAI 5

Resolution of SY-A18-01 [SR SY-A19 in the current version of the ASME/ANS Risk Standard] involves use of a systematic approach to consider maintenance unavailability, some of which may be overlapping, or not precluded by operating procedure limitations, which remains unresolved. The standard requires:

> In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened, in a manner consistent with the actual practices and history of the plant for removing equipment from service.

The possibility of partially overlapping component unavailability has not yet been resolved, but is in all cases conservative because component unavailability combinations that would normally not be possible are being added into the CDF and LERF differential quantifications as stated in the LAR. The disposition does not determine the extent of such overlapping unavailability, but rather a-priori assumes that, if there are, modeling would be less conservative than currently failing to model. Given that risk changes, before versus after, are the subject of concern, such conservatism, if applied to the before risk, could actually generate non-conservative risk increases when a larger risk is subtracted from the after risk than a more accurate smaller risk. Provide:

a. A further discussion of plant practices and the modeling of these practices relevant to overlapping simultaneous test and maintenance ™ unavailability.

b. A verification by example, or analysis, that this presumed conservatism is such that it ensures the differential risk for the application is also conservative (i.e., the risk estimated for the before versus after condition is not overestimated, such that subtracting it from the after value could underestimate the risk increase).

Exelon Response to RAI 5.a

A review of schedules and practices indicated that when two Functional Equipment Groups (FEGs) are scheduled in the same week at Ginna, the current practice is to sequence the FEGs rather than work them simultaneously. Exceptions to this practice are very rare and are carefully discussed, risk assessed, and unavailability recorded. This minimizes the concern of shadowing of maintenance unavailability. However, since overlap of these combinations is not procedurally excluded, coincident maintenance may occur and this is allowed via random combinations of maintenance events in the PRA model.

Exelon Response to RAI 5.b

As discussed in the response to RAI 5.a, the Ginna PRA model does not preclude overlapping maintenance of certain [SSCs] even though overlapping maintenance is not typically done at Ginna. However, certain overlapping maintenance configurations that are explicitly excluded by [TSs] are removed from cutsets through the use of the mutually

exclusive file. These typically include disallowed maintenance, such as both trains of a two-train TS system.

As such, the Ginna PRA model will include some cutsets with random combinations of maintenance configurations which will tend to increase the 'base' (average maintenance) CDF. However, higher risk combinations (such as both trains of Emergency Core Cooling Systems or both trains of Auxiliary Feed Water (AFW)) are precluded (through the mutually exclusive file). Therefore, the combinations of maintenance events that are in the cutsets but are not routinely entered should not be significant contributors to base CDF.

Additionally, test and maintenance (TM) basic event probabilities are calculated from all unavailability events, both planned and emergent. It is possible that configurations can occur in the plant due to one train being in planned maintenance and the failure of another train.

The Average Test and Maintenance (ATM) model base case results are conservative due to the possibility of cutsets which contain random overlapping maintenance unavailability events. This unavoidable conservatism in the base result however has little effect on the delta risk for this given application.

An example of this in the ATM model is the highest cutset with two TM terms that could potentially have a conservative effect. This cutset is shown in the table below.

CDF (Cutset)	**Basic Event Value**	**Event Name**	**Description**
$6.53x10^{-9}$	$3.65x10^{+2}$	TI0000SW	TOTAL LOSS OF SERVICE WATER
	$1.29x10^{-2}$	AFTMOTDAFW	[Turbine-driven AFW] TDAFW PUMP TRAIN OUT-OF-SERVICE FOR MAINTENANCE
	$1.60x10^{-2}$	AXHFR04084	[Standby AFW] SAFW TRAIN C FT-4084 RESTORATION ERROR AFTER CALIBRATION
	$1.03x10^{-2}$	AXTMSAFSGB	SAFW TRAIN D TO S/G O.O.S. DUE TO T/M
	$9.41x10^{-1}$	MODE1	MODE1
	$1.00x10^{0}$	NO SBO	TAG- NO STATION BLACKOUT (SBO)
	$8.94x10^{-6}$	SWCCFPUMPR_ALL	CCF OF ALL COMPONENTS IN GROUP 'SWCCFPUMPR'
	$1.00x10^{0}$	TRANSX	TAG- TRANSIENT EVENT

In the case of a hypothetical STI risk evaluation, suppose that the STI change involves the service water pumps such that the value of the SWCCFPUMPR_ALL basic event in the above cutset is increased consistent with the surveillance frequency change program methodology (e.g., by a factor of 2). For each STI evaluation, the delta risk is always driven by changes to a specific set of basic event values unique to a specific surveillance test. As such, any basic events that appear in cutsets with overlapping maintenance unavailability events would correspondingly increase for the STI evaluation. This would be conservative in all cases (i.e., the risk estimated for the before versus after condition is included in both cases, such that subtracting it from the after value will not underestimate the risk increase). For our hypothetical

case (factor of 2) the cutset value increases to 1.31x10-8, with a delta CDF of -6.53x10-9. This is conservative, yet is far below the acceptance criteria for STI changes showing the insignificant impact of the ATM model conservatism.

NRC Evaluation of Exelon Response to RAIs 5.a and 5.b

Based on the licensee's information and discussion to resolve the RAI and its subparts, the NRC staff finds this supporting requirement acceptable for use in making STI changes.

<u>RAI 6</u>

A PRA model regarding resolution of F&O IE-C13-01 [SR IE-C15 in the current version of the ASME/ANS Risk Standard] is required by the standard to do the following: "CHARACTERIZE the uncertainty in the IE frequencies and PROVIDE mean values for use in the quantification of the PRA results." The sources of uncertainty which are 'considered' versus 'not considered' in estimation of mean values of any cutset element form an important input in judging the technical adequacy of a PRA model. The original peer review noted that "Section 5 [of the IE Notebook] does not provide or reference the parametric uncertainty IE data distribution [with a specific example cited]." The LAR treatment of this F&O expresses an opinion that while this 'documentation only' is still unresolved, this issue would not impact TSTF-425 PRA evaluations. The sources of uncertainty that were actually considered are an integral part when assessing PRA technical adequacy. Therefore:

a. Characterize what types of uncertainties are actually considered in the estimation of each IE mean frequency in the current PRA model of record.

b. Clarify if this currently unresolved F&O IE-C13-01 was subsequently re-evaluated in the 2012 FPRA peer review as a "back-referenced" supporting requirements item.

Exelon Response to RAI 6.a

Assumptions and uncertainties are addressed in the Section 5.0 of the Ginna IE Notebook. This section addresses uncertainties such as only using industry [l]oss of [o]ffsite power data from 1997-2013 due to deregulation, and not Bayesian updating LOCA values as they are industry expert's best estimates.

Error Factors (EFs) were added to Table 4-1 in the IE Notebook as part of the 2015 PRA update. The EFs from the generic data are used as an input into the Bayes update process and updated accordingly with the plant-specific evidence. For SSIE fault tree quantification, uncertainty was captured at the basic event level. In some cases, split fractions were applied to generic [IE] frequencies. In these cases, the Jeffreys' non-informative prior alpha factor of 0.5 was used. The EFs were then estimated from the corresponding statistical distribution.

NRC Evaluation of Exelon Response to RAI 6.a

The licensee stated that the missing EFs noted in the peer review F&O were added to the IE Notebook in the 2015 PRA update. The licensee's stated approach for dealing with uncertainties in 'point estimate values' is the use of a Jeffreys' non-informative prior distribution, or: Beta (1/2, 1/2). This approach is consistent with the uncertainty analysis approach found in

Section 6.3.2.3.2 of NUREG/CR-6823, "Handbook of Parameter Estimation for Probabilistic Risk Assessment" [5], and is thus acceptable for use in the TSTF-425 program.

Exelon Response to RAI 6.b

This F&O was addressed in the Fire Uncertainty Notebook for the fire related initiating events. This Fire Uncertainty Notebook did not address non-fire related initiating events, but as discussed in response to RAI 6.a, this F&O has been addressed as part of the 2015 internal events PRA model update.

NRC Evaluation of Exelon Response to RAI 6.b

Based upon the licensee's use of an acceptable method for treating uncertainties in their 2015 PRA update, the approach is acceptable to fully resolve F&O IE-C13-01 for the TSTF-425 program.

RAI 7

Resolution of F&O HR-G3-01 was based upon conformance with Revision 1 of RG 1.200. The assessment of PRA Technical Adequacy must address conformance with Revision 2 of RG 1.200. A number of specific clarifications to the ASME/ANS Risk Standard regarding SA HR-G3 were added by Revision 2 of RG 1.200 and noted below:

Cat I:

(a) The complexity of detection, diagnosis, decision-making and executing the required response.

Cat II, and III:

(d) Degree of clarity of the cues/indications in supporting the detection, diagnosis, and decision-making give the plant-specific and scenario-specific context of the event.

(g) Complexity of detection, diagnosis and decision-making, and executing the required response.

Provide a gap assessment of the current Human Reliability Analysis (HRA) in the PRA model of record against the additional clarifications in Revision 2 of RG 1.200 noted above.

Exelon Response to RAI 7

This F&O was addressed in the Fire [HRA] Notebook for the fire related human actions. This included almost all of the non-fire related HRA events as most of the non-fire related HRA events are included in the fire model as well.

Consideration of cue clarity and complexity were considered as part of the 2015 internal events model update for Ginna. Any and all additions to cue clarity and complexity have been incorporated into the HRA Calculator database file for the FPIE model, and will also be

incorporated in Appendix I of the 2015 Ginna FPIE HRA Notebook. As such, the 2015 internal events PRA model update is consistent with HR-G3 including the clarifications provided in [Revision 2 of RG 1.200].

NRC Evaluation of Exelon Response to RAI 7

Based on the license's considerations and update to the internal events PRA model, the NRC staff finds the licensee's disposition of F&O HR-G3-01 to be acceptable for use in the TSTF-425 program.

RAI 8

Resolution of F&O IF-C8-01 [IFSN-A16 in the current version of the ASME/ANS Risk Standard], involves a flooding "source that was screened based on qualitative consideration of potential human action; but for that action (in response to a 2,000 gpm Fire Service Water (FSW) break in the Intermediate Building North (IBN)), there doesn't appear to be any justification for the time identified (190 min). Nothing other than time available is cited as rationale for screening the event." The LAR states that the "impact is expected to be minimal, and is not expected to have any impact on the [SFCP]." Without having corrected the PRA model of record to address the specific internal flood source issue it is not readily obvious how the conclusion of minimal impact was obtained. The licensee was requested to provide the technical bases for assuring this omitted flood source in fact does not have any impact on the TSTF-425 based SFCP.

Exelon Response to RAI 8

The FSW breaks in the IBN are no longer screened, since they are now being represented by the internal flood initiator FL-IBN-FSW-2K.

NRC Evaluation of Exelon Response to RAI 8

The licensee's response to RAI 8 indicates the omitted flooding source is now modelled in the Ginna PRA, which is acceptable for use in the TSTF-425 program.

RAI 9

The guidance for changing TS surveillance frequencies is provided in Revision 2 of RG 1.177. However, for allowable risk changes associated with surveillance frequency extensions, it refers to Revision 2 of RG 1.174 which provides quantitative risk acceptance guidelines for changes to CDF and LERF. Revision 2 of RG 1.200 is invoked by Revision 2 of RG 1.174 to address PRA Technical Adequacy. Portions of the ASME/ANS RA-Sa-2009 standard, with clarifications, is endorsed by Revision 2 of RG 1.200. The RITS-5b LAR is based upon Revision 3 of TSTF-425 and a PRA Model, which was assessed in a peer review for conformance with Revision 1 of RG 1.200. Conformance with the requirements in Revision 2 of RG 1.200 is a requirement. Therefore:

a. Provide a gap analysis to identify any areas where the current PRA model of record does not conform to the PRA Technical Adequacy requirements in Revision 2 of RG 1.200 and the ASME/ANS RA-Sa-2009 standard.

b. Clarify how the PRA applications associated with RITS-5b will not be impacted by the gaps in the PRA model conformance with Revision 2 of RG 1.200.

c. Clarify that there have been no PRA model upgrades as defined in Appendix 1-A of ASME/ANS RA-Sa-2009, which would require a focused peer review. Specifically, discuss whether the addition of two diesel generators as an alternate source of power to the SAFW pumps and a condensate storage tank as a dedicated water source for these pumps in model GN114A-W constitutes an upgrade. If so, has there been a focused-scope peer review? If not, justify.

d. Confirm that the total baseline risk is consistent with the quantitative risk acceptance guidelines in Revision 2 of RG 1.174, which provides for changes to CDF and LERF.

Exelon Response to RAI 9.a and 9.b

A gap assessment was performed for the internal events PRA between [Revision 1 of RG 1.200] and [Revision 2 of RG 1.200]. This gap assessment did not lead to the identification of any new "Not Mets" or changes to the original [CC] ranking from the 2009 peer review. The results of the 2011 gap assessment provided the origin for the dispositions provided in Table 2-1 [in] the LAR, which has subsequently been updated in the response to RAI 1 above. Therefore, the identification and disposition of the internal events model gaps provided in RAI 1 is consistent with the PRA Technical Adequacy requirements [in Revision 2 of RG 1.200] and the ASME/ANS RA-Sa-2009 standard.

NRC Evaluation of Exelon Response to RAI 9.a and 9.b

Based on the licensee's clarifications regarding the gap assessment, the NRC staff finds the information acceptable for use in the TSTF-425 program.

Response to RAI 9.c

The addition of the two diesel generators as an alternate source of power to the SAFW pumps and a condensate storage tank as a dedicated water source in the updated PRA model utilized methods consistent with the peer-reviewed PRA model. Additionally, other changes to the PRA model were also developed consistent with the methods employed in the peer-reviewed PRA model. As such a focused-scope peer review of the internal events PRA model is not currently warranted. However, as discussed in the response to RAI 2.c, Ginna plans on transitioning to Appendix L of NUREG/CR-6850 for determining revised MCB fire frequencies. This will require a focused scope peer review.

NRC Evaluation of Exelon Response to RAI 9.c

The licensee indicates that the additional hardware modifications will be modelled consistent with the technical approaches used in the existing Ginna PRA models and thus would not require a focused-scope peer review. The licensee also indicated they would be transitioning to Appendix L of NUREG/CR-6850 for determining revised MCB fire frequencies and that this would require a focused-scope peer review. The effects of these changes to the Ginna PRA model would then be utilized in future evaluations of TS surveillance frequency changes. Thus

the licensee's response to RAI 9.c provides the information and discussion to resolve the RAI and is acceptable for use in the TSTF-425 program.

Exelon Response to RAI 9.d

As provided in Attachment W of the NFPA-805 LAR submittal [7], the RG 1.174 guidelines are met. It should be noted that the internal event CDF and LERF values in the most recent version of the PRA model are lower than that reported in the NFPA-805 LAR submittal. It is understood that those guidelines must continue to be met to allow the use of risk-informed applications.

NRC Evaluation of Exelon Response to RAI 9.d

The licensee's response to RAI 9.d provides the information and discussion to resolve the RAI and is acceptable for the TSTF-425 program.

RAI 10

A significant model change is defined in Revision 2 of RG 1.200 as the following: "Whether a change is considered significant is dependent on the context in which the insights are used. A change in the risk insights is considered significant when it has the potential to change a decision being made using the PRA." F&Os IF-D5a-01 (unresolved), IF-07-01, [IFEV-A6, IEFV-A8 in the current version of the ASME/ANS Risk Standard], in the current PRA model of record, involve:

a. Not adequately addressing plant-specific characteristics that might affect the manner in which the frequencies of flooding are estimated (e.g., material condition, aging degradation, and water-hammer potential).

b. Inappropriate screening (out) of certain internal flood scenarios without applying consistent screening criteria, as required in SRs IF-07 and IF-E3a.

If the frequencies of specific internal floods are improperly evaluated, the importance of specific flood scenarios and how they impact the unavailability of specific components will be inappropriate, and this will impact the technical adequacy of the PRA model of record. The RITS-5b LAR indicates that a sensitivity evaluation for a particular STI evaluation will be performed to determine if there is any impact. Within the scope of Revision 3 of TSTF-425, clarify:

a. The specific sensitivity studies which are to be performed with the PRA model of record in order to demonstrate technical adequacy of the internal flooding frequencies without correcting the identified deficiency noted in the peer review F&O IF-D5a-01.

b. The impact on the unavailability of specific components evaluated in the SFCP of "screening in" internal flood sources which were eliminated in the current PRA model of record.

Exelon Response to first RAI 10.a

A review of plant-specific operating experience for Ginna determined that there were no significant flooding events that have occurred in the past 16 years (since August 1998). Based on this, the generic industry failure rates developed by [Electric Power and Research Institute] EPRI are acceptable for use, and a plant-specific update of these frequencies is not deemed warranted as a Bayes update with no events will have a very minimal impact on the flooding frequencies. This is documented in the latest revision of the Internal Flood Notebook (G1-PRA-012). The material condition and aging management strategies for plant piping are addressed via the Risk Informed In-Service Inspection programs that are implemented at the site. The effects of water hammer on plant piping are inherently included as a part of the calculated rupture frequencies developed by EPRI based on industry experience.

Exelon Response to first RAI 10.b

Screening criteria based on the ASME/ANS PRA Standard has now been consistently applied to flood sources and areas. This is documented in the latest revision of the Internal Flood Notebook (G1-PRA-012).

Exelon Response to second RAI 10.a

For each SFCP analysis, a review will be made to see if the adjusted basic events for the components or system of interest appear in cutsets concurrent with a particular flood initiator, of notable significance, e.g., greater than a 10% contribution to the calculated change in CDF or LERF. If so, a specific sensitivity analysis will be performed related to the flood frequency to see if it could influence the acceptability of the STI change evaluation consistent with the guidance in Step 14 of NEI 04-10 [6]. The potential need for this sensitivity will be controlled via the PRA model [URE] database that is reviewed any time the PRA model is used for a documented risk application.

Response to second RAI 10.b

The most recent PRA model update carefully considered the criteria in the ASME/ANS PRA Standard with regard to being able to screen flood areas and water sources. Based on the current PRA update, there are no longer any water sources that were inappropriately screened, and no scenarios were numerically screened using Supporting Requirement IFQU-A3 of the PRA Standard. Because of this, there is no risk of "screening in" any new internal flood scenarios.

NRC Evaluation of Exelon Response to RAI 10

The licensee's response to RAI 10 provides the information and discussion to resolve the RAI and is acceptable for the TSTF-425 program.

RAI 11

Similar to F&O IE-C13-01 dealing with internal events, Internal Flooding F&O IF-F3-01 [IFQU-83 in the current version of the ASME/ANS Risk Standard], and which is still unresolved, identified deficiencies in the consideration of uncertainties and that the treatment "did not constitute an

adequate characterization of the sources of uncertainty associated with the flood analysis or a comprehensive discussion of the assumptions that could have an effect on the results." The LAR treats this F&O as a "documentation only F&O" which will not impact evaluation of specific components in the SFCP. Without knowing what sources of uncertainty were actually considered, and how such uncertainties propagate to the end results, it was not possible for the original peer review to assess the required technical adequacy. Therefore:

a. Characterize what types of uncertainties are actually considered in the estimation of each IE mean frequency in the current PRA model of record.

b. Clarify if this currently unresolved F&O IF-F3-O1 was subsequently re-evaluated in the 2012 FPRA Peer Review as a "back-referenced" supporting requirements item.

Exelon Response to RAI 11.a

In estimating the event mean frequency for each internal flood initiator, the [IE] uncertainty parameters from the EPRI 1013141 data were used and [EFs] reported in the Internal Flood Notebook (G1-PRA-012). These parametric uncertainty values propagate to the end results using the CAFTA PRA software. Modeling uncertainty for the internal flood portion of the PRA was also addressed and documented in G1-PRA-012 using the guidance found in EPRI 1016737. The finding for IF-F3-01 is considered to be resolved.

Exelon Response to RAI 11.b

Per the response to RAI 11.a, IF-F3-01 is now considered to be resolved.

NRC Evaluation of Exelon Response to RAI 11

Based on the licensee's description of the uncertainty evaluation, the NRC staff finds this response to RAI 11 acceptable for use in the STI change evaluations.

RAI 12

F&O IF-E5-01 [IFQU-A5 in the current version of the ASME/ANS Risk Standard], involved use of HRA methods, which were not consistent with the methods used elsewhere in the PRA model. The LAR indicates the issue has been resolved. The Non-Mandatory Appendix 1-A of the ASME/ANS Risk Assessment Standard would require a focused peer review if there was an underlying PRA model upgrade (e.g., application of new methods which were different than those in the original model), but not for PRA model maintenance, where PRA model maintenance is specifically defined: "plant modifications, procedure changes, plant performance (data)." Confirm that the revised HRA performed for the internal flooding portion of the PRA model of record uses HRA methods that are consistent with other portions of the PRA that have been peer reviewed. If not, confirm whether a focused peer review had been performed for the internal flooding HRA consistent with the requirements in Appendix 1-A of ASME/ANS RA-Sa-2009.

Exelon Response to RAI 12

As discussed under PRA RAI 7 related to F&O HR-G3-01 an improved HRA method was implemented as part of the fire analysis. This method was peer reviewed as part of the fire evaluation.

The internal flooding HRA document was reviewed to ensure consistency between the HRA methodology applied in the analyses of both internal flooding and FPIE operator actions. The internal flooding analyses were determined to be consistent with the methodology applied throughout the FPIE HRA. Additionally, the internal flood [Human Failure Event] analyses were included into the HRA Calculator database to ensure consistency in future updates of the FPIE HRA. The results of these analyses are included in the 2015 Ginna FPIE HRA Notebook.

NRC Evaluation of Exelon Response to RAI 12

The NRC staff finds the licensee's response to RAI 12 acceptable for use in the TSTF-425 program because of its description of the consistency evaluation for the internal events PRA methodology.

RAI 13

In Section 2.0.5 in Attachment 2 of the LAR, it is stated that:

> The results of the standby failure rate sensitivity study plus the results of any additional sensitivity studies identified during the performance of the reviews as outlined in 2.2.1 and 2.2.3 above for each STI change assessment will be documented and included in the results of the risk analysis that goes to the Integrated Decision-making Panel (IDP).

The LAR does not contain any Sections 2.2.1 or 2.2.3. Correct the LAR to address the missing Sections 2.2.1 and 2.2.3.

Exelon Response to RAI 13

The LAR does not contain any missing sections. The paragraph quoted above was submitted with a typographical error. Specifically, the reference to sections 2.2.1 and 2.2.3 should have read 2.0.2 and 2.0.4, respectively.

NRC Evaluation of Exelon Response to RAI 13

The licensee's response to RAI 13 indicates the problem was caused by a typographical error. Thus the response provides the corrected information to resolve the RAI.

RAI 14

The LAR states in Section 2.0.4 of Attachment 2 with regard to the most recent PRA model GN114A-W and peer reviews conducted for the internal events model in 2009 and FPRA model in 2012:

> All remaining gaps will be reviewed for consideration during the 2015 model update but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

Confirm that any gap assessment and, if identified as required due to model upgrades, focused-scope or full-scope peer review will be performed in accordance with the then latest version of the ASME/ANS PRA Standard as endorsed, clarified and qualified by Revision 2 of RG 1.200.

Exelon Response to RAI 14

The current status of the gaps to [Revision 2 of RG 1.200] based on the most recent internal events PRA model update are provided in response to RAI 1. As noted in response to RAI 9, other changes to the PRA model were also developed consistent with the methods employed in the peer-reviewed PRA model. As such, a focused-scope peer review of the internal events PRA model is not currently warranted.

NRC Evaluation of Exelon Response to RAI 14

The licensee's discussion of its rationale behind not conducting any further gap assessments or peer reviews prior to implementation of the TSTF-425 is acceptable and will depend on model changes going forward.

RAI 15

The need for realistic treatment of feasible operation action after core damage was addressed in F&O LE-C2a-01, noting it is conservative not to credit these. The cited impact to TSTF-425 stated that there are limited operator actions that could influence LERF, such that their effect is unlikely to be significant, possibly even lowering LERF estimates. Therefore, the omission of these actions is conservative and does not adversely impact the PRA model used for TSTF-425 analysis.

Conservatism in the before versus after risk when performing a risk increase calculation does not guarantee a conservative estimate of the risk increase, since a more realistic estimate of the before risk, being lower, would lead to a more conservative estimate of the risk increase when before is subtracted from after. The licensee was requested to either demonstrate essentially no effect on the before risk by excluding credit for these actions or reassess the before risk, and therefore the risk increase, after incorporating credit for these actions.

This was supplemented by the following: Was the cited LERF assessment based upon the Pressurized Water Reactor Owners Group (PWROG) Simplified LERF Methodology contained in WCAP-16341-P? Additionally, in modelling human recovery actions (e.g., the two cited in the response - late restoration of offsite power and late reactor coolant system (RCS) depressurization), do any of these actions depend on the functioning of active components which are subject to changes in TS STIs such as, but not limited to, batteries, electrical breakers, power-operated relief valves (PORVs), and instrument air systems?

Exelon Response to RAI 15

Two human actions are identified in the Level 2 analysis that may be credited in the LERF PRA model for human action post-core damage, but prior to vessel breach: (1) late recovery of offsite power in station blackout scenarios where core damage is arrested prior to vessel breach and (2) late depressurization of the [RCS].

Late recovery of offsite power is explicitly modeled in the LERF PRA. The Ginna PRA LERF assessment utilizes the methodology set forth in the PWROG document WCAP-16341-P, Simplified Level 2 Modeling.

In the Ginna Level 2 Analysis, the probability of an early Containment failure is dependent on the loads on the Containment at vessel breach. One factor that can affect Containment loads is [RCS] pressure at vessel breach. RCS depressurization prior to core damage is credited in the PRA LERF model. Given that the RCS is not depressurized early, a late depressurization action is feasible. However, the system responses that would be measured by a STI change are already credited in the early depressurization action. Failure of those systems early would fail the late action and would not non-conservatively impact the delta risk calculation. In addition, in the LERF accident progression, the late RCS depressurization action would only impact containment failure probabilities.

An [URE] is open to develop and implement a human error probability for the late depressurization action. As with all TSTF-425 related assessments, the delta risk results will be reviewed to ensure that no conservatisms are significantly masking the delta risk evaluation.

The modeling for recovery actions for "early [RCS] depressurization" includes the hardware and support system functions, such as [PORVs]. The modeling for recovery actions for "late RCS depressurization" does not impact LERF results, and does not impact the risk assessment for [STI] analysis.

Offsite power recovery is credited in both the Level 1 (CDF) accident sequences and the Level 2 (LERF) accident sequence for station blackout events. Offsite power is supplied to the plant through the 115 KV Switchyard Station 13A, which is located outside of the protected area. No breakers located in Switchyard Station 13A are within the scope of the [SFCP].

Component basic events to recover power to the site's safety-related 480V busses are modeled in the Level 1 (CDF) accident sequences but not in the Level 2 (LERF) accident sequences. The risk significance of the LERF power recovery event is very low. Some components between Station 13A and the safety-related 480V busses are in the scope of the Surveillance Frequency Change Program.

A[n] URE has been initiated to add the offsite power recovery component basic events to the Level 2 model. Until the URE item is closed, the URE will be reviewed for applicability for each [STI] change evaluation as required by Exelon procedure guidance.

NRC Evaluation of Exelon Response to RAI 15

The licensee's response to RAI 15 identifies two human actions in the Level 2 analysis that may be credited in the LERF PRA model for human action post-core damage, but prior to vessel

breach. The licensee further clarifies that late recovery of offsite power is explicitly modeled in the LERF PRA, which utilizes the methodology set forth in the PWROG document WCAP-16341-P, Simplified Level 2 Modeling. Additionally, the modeling for recovery actions for "late RCS depressurization" does not impact LERF results, and does not impact the risk assessment for STI analysis. Based on the information and discussion provided by the licensee response, the NRC staff finds the response acceptable for use in the TSTF-425 program.

RAI 16

Survivability credit for equipment or human actions that could be impacted by containment failure was addressed in F&O LE-C9a-01; stating that it did not appear such credit was taken, leaving this supporting requirement as CC-I, acknowledged as not applicable in the disposition and impact on TSTF-425. If crediting equipment survivability in the before versus after risk condition would lead to a more conservative estimate of the risk increase, then it may not only be non-conservative to have ignored this, but also may fail to meet even CC-I for the application where it is the risk increase that is the key, not the base risk. Further, not applicable may not be an appropriate disposition. The licensee was requested to address this F&O in light of the potential effect on risk increase, not only base risk, with regard to TSTF-425.

Exelon Response to RAI 16

In the Ginna Level 2 Analysis, early containment failure after core-damage and vessel breach is the end-state for the LERF accident progression. There are no equipment dependencies or human actions that are identified that could be reasonably credited to prevent a release through a failed containment. There is no credited equipment, systems, or human actions that would be impacted by the adverse environment impacted by containment failure. Therefore, this issue would not impact delta-risk calculations. A[n] URE is open to capture that this F&O will remain unresolved and the supporting requirements will remain Category I.

NRC Evaluation of Exelon Response to RAI 16

The licensee stated that the Ginna Simplified LERF PRA model identified no equipment dependencies or human actions that could be credited to prevent a large early release through a failed containment. The licensee's response to RAI 16 thus provides the information and discussion necessary to resolve the RAI, and is acceptable for use in the TSTF-425 program.

RAI 17

Realistic containment bypass analysis was addressed by F&O LE-C10-01, including justification for any scrubbing credit, stating that no such credit was taken, although there was a sensitivity analysis determining any impact would be negligible. As a result, no impact on TSTF-425 was cited.

Verify that the impact of not considering scrubbing is negligible with respect to the risk increase from the before versus after risk calculation, not just negligible with respect to the base risk.

Exelon Response to RAI 17

In the Ginna Level 2 analysis, no credit is given for scrubbing of release paths. However, the Ginna Level 2 analysis identifies that scrubbing may be applicable to the following three

containment bypass conditions: (1) a steam generator tube rupture event with [AFW] available, or (2) internal flood scenarios with an interfacing system LOCA and the affected auxiliary building room flooded, or (3) sequences where the interfacing system LOCA break is in the RHR pits, thus resulting in the break potentially being submerged under a substantial water level.

NRC Evaluation of Exelon Response to RAI 17

The licensee responded that the only scenario where a possibly credited system could result in scrubbing was the availability of AFW for a tube rupture event. Other LERF sequences involving passive scrubbing of interfacing system LOCA scenarios would not be affected by changes in TS surveillance frequencies.

Based on the licensee's assessments using the currently applicable PRA standard and revision of RG 1.200, the NRC staff concludes that the level of PRA quality, combined with the evaluation and disposition of gaps, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 in Revision 1 of RG 1.177.

3.1.4.2 Scope of the PRA

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

The licensee has an at-power internal events and internal flooding PRA model as well as an at-power FPRA, to support the adoption of NFPA-805. Ginna has submitted a LAR for conversion from Appendix R compliance to NFPA-805 for fire protection which has been approved. Pursuant to this change, a FPRA has been created and implemented at Ginna. This FPRA model was created under the auspices of NUREG/CR–6850 and has undergone PWROG peer review (completed August 2012). The Ginna FPRA was developed using the National Institute of Standards and Technology (NIST) Consolidated Model of Fire and Smoke Transport Methodology; the Fire Dynamics Simulator, also developed by NIST; NUREG–1805 Fire Dynamics Tools computational spreadsheets; EPRI/NRC-RES FPRA Methodology for Nuclear Power Facilities and the associated NUREG/CR–6850 Frequently Asked Questions Process; Fire Events Database and plant specific data. This FPRA has numerous capabilities not considered in the Individual Plant Examination of External Events (IPEEE) FPRA model including explicit analysis of all risk significant fire areas such as the MCR and Relay Room. Multiple spurious operation considerations are also included. The ignition frequencies for all fire areas were developed using the guidance in NUREG/CR–6850 and also incorporate revised guidance for ignition frequencies. In accordance with Revision 1 of NEI 04-10 the licensee will use these models to perform quantitative evaluations to support the development of changes to surveillance frequencies in the SFCP. This is acceptable because the NRC-approved methodology in Revision 1 of NEI 04-10 allows for more refined analysis to be performed supporting changes to surveillance frequencies in the SFCP.

The licensee stated, in Attachment 2 of their application dated June 4, 2015, that external hazards were evaluated in the Ginna IPEEE submittal in response to the NRC IPEEE Program (Supplement 4 of Generic Letter 88-20). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks. The primary areas of external event evaluation at GINNA were internal fires and seismic risk. The internal fire events were addressed by using a combination of the EPRI Fire Induced Vulnerability Evaluation methodology and fire PSA. The results of the Fire Analysis are documented in the Ginna Nuclear Power Plant IPEEE Fire Analysis transmitted to NRC in June 1998. The seismic evaluations were performed in accordance with Generic Implementation Procedure (GIP) developed by the Seismic Qualification Utility Group of which Ginna was a member. The GIP provided plants a method for addressing Unresolved Safety Issue A-46 (Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors (USI A-46). Beyond this, Ginna performed a reduced-scope IPEEE for seismic events to close out IPEEE for Seismic Events. The Ginna USI A-46 Seismic Evaluation Report and the IPEEE Seismic Evaluation Report were transmitted to NRC in January 1997. However, there are no comprehensive CDF and LERF values available from the seismic IPEEE report to support the STI risk assessments. This supports the licensee's conclusion that insights from the IPEEE seismic PRA can be used to support the SFCP. This is an acceptable approach in accordance with Revision 1 of NEI 04-10.

The licensee stated that high winds, external floods, and transportation accidents were reviewed against the SRP as Ginna was one of the 11 participants in the NRC's Systematic Evaluation Program (SEP). Following plant modifications, it was determined that the Ginna plant met the SRP criteria. Based on the NRC SE Reports for Ginna's SEP results, no further submittals for Supplement 4 to GL 88-20 were warranted for high winds, external floods, or transportation accidents. This is an acceptable approach in accordance with Revision 1 of NEI 04-10.

Thus, the staff concludes that through the application of NRC-approved Revision 1 of NEI 04-10, the licensee's evaluation methodology is sufficient to ensure the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation and is consistent with Regulatory Position 2.3.2 in Revision 1 of RG 1.177.

3.1.4.3 PRA Modeling

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

Thus, the staff concludes that through the application of NRC-approved Revision 1 of NEI 04-10, the Ginna PRA modeling is sufficient to ensure an acceptable evaluation of risk for

the proposed changes in surveillance frequency, and is consistent with Regulatory Position 2.3.3 in Revision 1 of RG 1.177.

3.1.4.4 Assumptions for Time Related Failure Contributions

The failure probabilities of SSCs modeled in PRAs may include a standby time-related contribution and a cyclic demand-related contribution. The time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency is adjusted by the criteria in Revision 1 of NEI 04-10. This is consistent with Section 2.3.3 in Revision 1 of RG 1.177, which permits separation of the failure rate contributions into demand and standby for evaluation of SRs. If the available data do not support distinguishing between the time-related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate (per unit time) is assumed to be unaffected by the change in test frequency, such that the failure probability is assumed to increase linearly with time, and will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The NEI 04-10 process requires consideration of qualitative sources of information with regards to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus, the process is not reliant upon risk analyses as the sole basis for the proposed changes.

The potential benefits of 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. 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 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 which may be required by the Maintenance Rule. Thus, the staff concludes that through the application of NRC-approved Revision 1 of NEI 04-10 the licensee has employed reasonable assumptions with regard to extensions of STIs, and is consistent with Regulatory Position 2.3.4 in Revision 1 of RG 1.177.

3.1.4.5 Sensitivity and Uncertainty Analyses

By having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with Revision 1 of NEI 04 10, the licensee will be required to have sensitivity studies that 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 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. The licensee will also be required to perform monitoring and feedback of SSC performance, once the revised surveillance frequencies is implemented. Thus, the staff concludes that through the application of NRC-approved Revision 1 of NEI 04-10, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key

assumptions and model limitations and is consistent with Regulatory Position 2.3.5 in Revision 1 of RG 1.177.

3.1.4.6 Acceptance Guidelines

The licensee will be required to 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 Revision 1 of NEI 04-10, in accordance with the TS SFCP. Each individual change to surveillance frequency must show a risk impact below 1×10^{-6} per year for change to CDF, and below 1×10^{-7} per year for change to LERF. These changes to CDF and LERF are consistent with the acceptance criteria in Revision 2 of RG 1.174 for very small changes in risk. Where the acceptance criteria in Revision 2 of RG 1.174 are not met, the process in Revision 1 of NEI 04-10 either considers revised surveillance frequencies, which are consistent with Revision 2 of RG 1.174, or the process terminates without permitting the proposed changes. Where quantitative results are unavailable for comparison with the acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or zero. Otherwise, bounding quantitative analyses are required which demonstrate the risk impact is at least one order of magnitude lower than the acceptance guidelines in Revision 2 of RG 1.174 for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact less than 1×10^{-5} per year for change to CDF, and less than 1×10^{-6} per year for change to LERF, and the total CDF and total LERF must be reasonably shown to be less than 1×10^{-4} per year and 1×10^{-5} per year, respectively. These values are consistent with the acceptance criteria in Revision 2 of RG 1.174, as referenced by Revision 1 of RG 1.177 for changes to surveillance frequencies.

Consistent with the NRC's SE dated September 19, 2007, for Revision 1 of NEI 04-10, 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 NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with insignificant risk increases (i.e., less than 5×10^{-8} CDF and 5×10^{-9} LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance in Revision 2 of RG 1.174 is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history. The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results. Post implementation performance monitoring and feedback are also required to assure continued reliability of the components. The licensee's application of NRC-approved Revision 1 of NEI 04-10 provides acceptable methods for evaluating the risk increase associated with proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 in Revision 1 of RG 1.177. Therefore, the staff concludes that the proposed methodology satisfies

the fourth key safety principle in Revision 1 of RG 1.177, by assuring any increase in risk is small consistent with the intent of the Commission's Safety Goal Policy Statement.

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

The licensee's adoption of Revision 3 of TSTF-425 requires application of Revision 1 of NEI 04-10 in the SFCP. 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 is required by Revision 1 of NEI 04-10. The monitoring and feedback includes consideration of Maintenance Rule monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions which may be required by the Maintenance Rule. The performance monitoring and feedback specified in Revision 1 of NEI 04-10 is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 in Revision 1 of RG 1.177. Thus, the staff concludes that the fifth key safety principle in Revision 1 of RG 1.177 is satisfied.

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

The licensee proposed including the SFCP and specific requirements into Section 5.5.17 of the Ginna TSs, as follows:

Surveillance Frequency Control Program

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

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

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

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

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

3.3 Deviations from TSTF-425 and Other Changes

The licensee states in Section 2.2 of the LAR that:

> The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3;

Additionally, Section 2.2, "Optional Changes and Variations" of the LAR cover the following: (material edited for clarity):

> Exelon Generation proposes variations or deviations from TSTF-425, as identified below, which includes differing Surveillance numbers.
>
> Revised (clean) TS pages are not included in this amendment request given the number of TS pages affected, the straightforward nature of the proposed changes, and outstanding Ginna amendment requests that will impact some of the same TS pages. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," in that the mark-ups fully describe the changes desired.
>
> This is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's Model Safety Evaluation published in the same Federal Register Notice. As a result of this deviation, the contents and numbering of the attachments for this amendment request differ from the attachments specified in the NRC staff's model application.
>
> Surveillance frequencies that have not been changed under the Surveillance Frequency Control Program (SFCP) may not be based on operating experience, equipment reliability or plant risk. Therefore, the TSTF and the NRC agreed that the TSTF-425 TS Bases insert, "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program," should be revised to state, "The Surveillance Frequency is controlled under the Surveillance Frequency Control Program." The existing TS Bases information will be relocated to the licensee-controlled SFCP.
>
> Attachment 5 provides a cross-reference between TSTF-425 versus the Ginna Surveillances included in this amendment request. Attachment 5 includes a summary description of the referenced TSTF-425 TS Surveillances, which is provided for information purposes only and is not intended to be a verbatim description of the TS Surveillances.
>
> This cross-reference highlights the following:
>
> a. Surveillances included in TSTF-425 and corresponding Ginna Surveillances have differing Surveillances numbers,

b. Surveillances included in TSTF-425 that are not contained in the Ginna TS,

c. Ginna plant-specific Surveillances that are not contained in TSTF-425 Surveillances and, therefore, are not included in the TSTF-425 mark-ups.

In addition, there are Surveillances contained in TSTF-425 that are not contained in the Ginna TS. Therefore, the NUREG–1431 mark-ups included in TSTF-425 for these Surveillances are not applicable to Ginna. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's Model Safety Evaluation dated July 6, 2009 (74 FR 31996).

Ginna TSs include plant-specific Surveillances that are not contained in NUREG-1431 and, therefore, are not included in the NUREG–1431 mark-ups provided in TSTF-425. Exelon has determined that the relocation of the Frequencies for these Ginna plant specific Surveillances is consistent with TSTF-425, Revision 3, and with the NRC staff's Model Safety Evaluation dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the Model Safety Evaluation.

Changes to the Frequencies for these plant-specific Surveillances would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require that Surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

Changes to Frequencies in the SFCP would be evaluated using the methodology and probabilistic risk guidelines contained in NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. ML071360456), as approved by NRC letter dated September 19, 2007 (ADAMS Accession No. ML072570267).

The NEI 04-10, Revision 1, methodology includes qualitative considerations, risk analyses, sensitivity studies and bounding analyses, as necessary, and recommended monitoring of the performance of systems, components, and structures (SSCs) for which Frequencies are changed to assure that reduced testing does not adversely impact the SSCs. In addition, the NEI 04-10, Revision 1 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176), relative to changes in Surveillance Frequencies.

Therefore, the proposed relocation of the Ginna plant-specific Surveillance Frequencies is consistent with TSTF-425 and with the NRC staff's Model Safety Evaluation dated July 6, 2009 (74 FR 31996).

As stated by the licensee, and confirmed by the staff, "The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3." The licensee further describes "Optional Changes and Variations" and cites the methodology from Revision 1 of NEI 04-10 as the means by which changes to frequencies in the SFCP would be evaluated. Since the optional changes and variations would use the methodology in Revision 1 of NEI 04-10, the Staff concludes they are acceptable.

3.4 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a licensee-controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the Administrative Controls of TSs. This amendment does not relocate surveillance frequencies that (1) reference other approved programs for the specific interval, (2) are purely event-driven, (3) are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs, or (4) are related to specific conditions. Revision 1 of NEI 04-10 is referenced by the SFCP and Subsection 5.5.17 in the TSs, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TSs to a licensee-controlled document, provided those frequencies are changed in accordance with the NRC-approved Revision 1 of NEI 04-10, which is specified in the administrative controls of the TSs.

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

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

Paragraph 50.36(c) of 10 CFR discusses the categories that will be included in TSs. Paragraph 50.36(c)(3) of 10 CFR discusses the specific category of supporting requirements and states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff finds that with the proposed relocation of surveillance frequencies to a licensee-controlled document and administratively controlled in accordance with the TS SFCP, the licensee continues to meet the requirements in 10 CFR 50.36.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of New York of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (80 FR 61482, dated October 13, 2015). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. James Barstow, Exelon to USNRC, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," June 4, 2015, ADAMS Accession No. ML15166A075.

2. David Gudger, Exelon, to USNRC, "Response to Request for Additional Information – Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," February 3, 2016, ADAMS Accession No. ML16034A139.

3. David Gudger, Exelon, to USNRC, "Supplemental Response to Request for Additional Information – Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," March 29, 2016, ADAMS Accession No. 16089A425.

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

5. Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF [Risk-Informed TSTF] Initiative 5b," ADAMS Accession No. ML090850642.

6. USNRC to Bryan Hanson, Exelon, "Request for Additional Information Regarding: Risk-Informed Technical Specifications Initiative 5B," ADAMS Accession No. ML15344A353.

7. USNRC Topical Report NEI 04-10, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, Risk-informed Technical Specification Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," September 19, 2007, ADAMS Accession No. ML072570267

8. USNRC Regulatory Guide (RG) 1.174, Revison 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," ADAMS Accession No. ML100910006.

9. USNRC RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," ADAMS Accession No. ML100910008.

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Principal Contributor: Raymond Gallucci

Date: June 28, 2016.

June 28, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear Operations, Inc.
R. E. Ginna Nuclear Power Plant
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: LICENSE AMENDMENT REQUEST TO RELOCATE SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE CONTROLLED PROGRAM – ADOPTION OF TSTF-425, REVISION 3 (CAC NO. MF6358)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment No. 122 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant (Ginna). This amendment is in response to your application dated June 4, 2015, as supplemented by letters dated February 3, 2016, March 29, 2016, and June 16, 2016. Exelon Generation Company, LLC submitted a license amendment request to change the Technical Specifications (TSs) by relocating specific TS surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force - 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force Initiative 5b". Additionally, the change added a new program, the Surveillance Frequency Control Program, to TS Section 5, Administrative Controls.

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

Sincerely,
/RA/
Diane Render, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 122 to Renewed License No. DPR-18
2. Safety Evaluation

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Amendment No.: ML16125A485 *by letter/email

OFFICE	LPLI-1\PM	LPLI-1\LA	DRA/APLA	DSS/STSB
NAME	DRender	KGoldstein	SRosenberg*	RElliott*
DATE	05/04/2016	05/10/2016	06/23/2016	03/02/2016
OFFICE	DE/EEEB	DE/EEEB	DE/EEEB	DE/EEEB
NAME	TMartinez (non-concur)	SSom (non-concur)	SRay (non-concur)	GMatharu (non-concur)
DATE	06/07/2016	06/07/2016	06/07/2016	06/07/2016
OFFICE	DE/EEEB	DE/EEEB	DE/EEEB	OGC
NAME	RFitzpatrick (non-concur)	RMathew (non-concur)	JZimmerman (non-concur)	JLindell (NLO w/noted changes)*
DATE	06/07/2016	06/07/2016	06/17/2016	06/15/2016
OFFICE	LPLI-1/BC	LPLI-1\PM		
NAME	RGuzman for TTate	DRender		
DATE	06/28/16	6/28/16		

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