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

January 12, 2011

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Generation Company
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: REQUEST TO RELOCATE SURVEILLANCE FREQUENCIES TO A LICENSEE-CONTROLLED PROGRAM (TAC NO. ME3587)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 274 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated March 24, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100840205), as supplemented by letters dated July 29, 2010 (ADAMS Accession No. ML102110459), and September 27, 2010 (ADAMS Accession No. ML102700481).

The changes revise the TMI-1 Technical Specifications (TSs) to relocate certain surveillance frequencies to a licensee-controlled program through the implementation of Nuclear Energy Institute 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." The changes are consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force Initiative 5b," Revision 3.

By letter dated May 27, 2010 (ADAMS Accession No. ML092740791), the NRC approved Amendment No. 273 to the TMI-1 Renewed Facility Operating License No. DPR-50. The NRC understands that this amendment will not be implemented at TMI-1 until the fall 2011 refueling outage. Therefore, based upon the required implementation date specified in the enclosure to this letter, Amendment 274 will be implemented at TMI-1 prior to Amendment 273. Amendment No. 273 changes TS pages 4-3 and 4-5, which are also changed by Amendment No. 274. To avoid operational confusion, the changes to pages 4-3 and 4-5 in the enclosed amendment are being issued such that they do not reflect the plant configuration changes covered by Amendment No. 273 because it is not yet implemented. A correction letter for pages 4-3 and 4-5 will be issued, under separate cover, to reflect the changes of both amendments such that the TSs will be in their proper configuration upon implementation of Amendment No. 273.

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.

Please contact me at 301-415-2833 if you have any questions.

Sincerely,

Peter Bamford

Peter J. Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 274 to DPR-50
2. Safety Evaluation

cc: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 274
License No. DPR-50

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated March 24, 2010, supplemented by letters dated July 29, 2010, and September 27, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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-50 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.274 , are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: January 12, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 274

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove	Insert
Page 4	Page 4

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	Remove	Insert
v	v	4-29	4-29
3-34a	3-34a	4-39	4-39
3-35a	3-35a	4-41	4-41
3-59	3-59	4-42	4-42
4-2	4-2	4-43	4-43
4-3	4-3	4-44	4-44
4-4	4-4	4-45	4-45
4-5	4-5	4-46	4-46
4-5a	4-5a	4-47	4-47
4-6	4-6	4-52	4-52
4-7	4-7	4-52a	4-52a
4-7a	4-7a	4-54	4-54
4-8	4-8	4-55	4-55
4-9	4-9	4-55f	4-55f
4-10	4-10	4-59	4-59
4-10a	4-10a	4-86	4-86
4-10b	4-10b	**	6-30
4-10c	4-10c		

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.274 are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Three Mile Island Nuclear Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

(4) Fire Protection

Exelon Generation Company shall implement and maintain in effect all provisions of the Fire Protection Program as described in the Updated FSAR for TMI-1.

Changes may be made to the Fire Protection Program without prior approval by the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided that interim compensate measures are implemented.

(5) The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to measure the values of the critical parameters;

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

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2. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt, in excess of the tilt limit, or when thermal power is equal to or less than 50% full power with four reactor coolant pumps running, set the nuclear overpower trip setpoint equal to or less than 60% full power.

3. The control rod group withdrawal limits in the CORE OPERATING LIMITS REPORT shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

4. The operational imbalance limits in the CORE OPERATING LIMITS REPORT shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

f. Except for physics or diagnostic testing, if quadrant tilt is in excess of the maximum tilt limit defined in the CORE OPERATING LIMITS REPORT and using the applicable detector system defined in 3.5.2.4.a, b, and c above, reduce thermal power to ≤15% FP within 2 hours. Diagnostic testing during power operation with a quadrant tilt is permitted provided that the thermal power allowable is restricted as stated in 3.5.2.4.d above.

g. Quadrant tilt shall be monitored on a minimum frequency of once every 12 hours when the QPT alarm is inoperable and at the frequency specified in the Surveillance Frequency Control Program when the alarm is operable during power operation above 15 percent of rated power. When QPT has been restored to ≤ steady state limit, verify hourly for 12 consecutive hours, or until verified acceptable at ≥95% FP.

e. If an acceptable axial power imbalance is not achieved within 24 hours, reactor power shall be reduced to ≤40% FP within 2 hours.

f. Axial power imbalance shall be monitored at the frequency specified in the Surveillance Frequency Control Program when axial power imbalance alarm is OPERABLE, and every 1 hour when imbalance alarm is inoperable during power operation above 40 percent of rated power.

3.5.2.8 A power map shall be taken at intervals not to exceed 31 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in the CORE OPERATING LIMITS REPORT.

Bases

The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. The effects of the APSRs are included in the limit development. Additional conservatism included in the limit development is introduced by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors

e. Postulated fuel rod bow effects

f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

The incore instrumentation system uncertainties used to develop the axial power imbalance and quadrant tilt limits accounted for various combinations of invalid Self Powered Neutron Detector (SPND) signals. If the number of valid SPND signals falls below that used in the uncertainty analysis, then another system shall be used for monitoring axial power imbalance and/or quadrant tilt.

For axial power imbalance and quadrant power tilt measurements using the incore detector system, the minimum incore detector system consists of OPERABLE detectors configured as follows:

Axial Power Imbalance

a. Three detectors in each of three strings shall lie in the same axial plane with one plane in each axial core half.

b. The axial planes in each core half shall be symmetrical about the core mid-planes.

c. The detectors shall not have radial symmetry.

Quadrant Power Tilt

a. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

3.14 FLOOD

3.14.1 PERIODIC INSPECTION OF THE DIKES AROUND TMI

Applicability

Applies to inspection of the dikes surrounding the site.

Objective

To specify the minimum frequency for inspection of the dikes and to define the flood stage after which the dikes will be inspected.

Specification

3.14.1.1 The dikes shall be inspected at the frequency specified in the Surveillance Frequency Control Program and after the river has returned to normal, following the condition defined below:

a. The level of the Susquehanna River exceeds flood stage; flood stage is defined as elevation 307 feet at the Susquehanna River Gage at Harrisburg.

Bases

The earth dikes are compacted to provide a stable impervious embankment that protects the site from inundation during the design flood of 1,100,000 cfs. The rip-rap, provided to protect the dikes from wave action and the flow of the river, continues downward into natural ground for a minimum depth of two feet to prevent undermining of the dike (References 1 and 2).

Periodic inspection, and inspection of the dikes and rip-rap after the river has returned to normal from flood stage, will assure proper maintenance of the dikes, thus assuring protection of the site during the design flood.

References

(1) UFSAR, Section 2.6.5 - "Design of Hydraulic Facilities"

(2) UFSAR, Figure 2.6-17 - "Typical Dike Section"

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The type of surveillance required for reactor protection system, engineered safety feature protection system, and heat sink protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1. The frequency of surveillance required for the instrumentation shown in Table 4.1-1 is specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2, 4.1-3, and 4.1-5 at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Tables 4.1-2, 4.1-3, and 4.1-5.

4.1.3 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.1-4.

4.1.4 Each remote shutdown system function shown in Table 3.5-4 shall be demonstrated OPERABLE by the performance of the following check, test, and calibration at the frequencies specified in the Surveillance Frequency Control Program:

a) Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.

b) Verify each required control circuit and transfer switch is capable of performing the intended function.

c) Perform CHANNEL CALIBRATION for each required instrumentation channel (excludes source range flux).

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. The acceptance criteria for the daily check of the Makeup Tank pressure instrument will be maintained within the error used to develop the plant operating limit. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated in the Surveillance Frequency Control Program is deemed adequate for reactor system instrumentation.

TABLE 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
1. Protection Channel Coincidence Logic	NA		NA	
2. Control Rod Drive Trip Breaker and Regulating Rod Power SCRs	NA		NA	(1) Includes independent testing of shunt trip and undervoltage trip features.
3. Power Range Amplifier	(1)	NA	(2)	(1) When reactor power is greater than 15%. (2) When above 15% reactor power run a heat balance check at the frequency specified in the Surveillance Frequency Control Program. Heat balance calibration shall be performed whenever heat balance exceeds indicated neutron power by more than two percent.
4. Power Range Channel			(1)(2)	(1) When reactor power is greater than 60% verify imbalance using incore instrumentation. (2) When above 15% reactor power calculate axial offset upper and lower chambers after each startup if not done within the previous seven days.
5. Intermediate Range Channel	(1)		NA	(1) When in service.
6. Source Range Channel	(1)		NA	(1) When in service.
7. Reactor Coolant Temperature Channel				

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
8. High Reactor Coolant Pressure Channel				
9. Low Reactor Coolant Pressure Channel				
10. Flux-Reactor Coolant Flow Comparator				
11. Reactor Coolant Pressure-Temperature Comparator				See Notes (a) and (b).
12. Pump Flux Comparator				
13. High Reactor Building Pressure Channel				
14. High Pressure Injection Logic Channels	NA		NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	(1)			(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
16. Low Pressure Injection Logic Channel	NA		NA	
17. Low Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	(1)			(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA		NA	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
b. RCS Pressure 1600 psig	(1)	(1)	NA	(1) When RCS Pressure > 1800 psig.
c. Deleted				
d. Reactor Bldg. 30 psi pressure switches	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	(1)	(1)(2)		(1) When CONTAINMENT INTEGRITY is required.
f. Line Break Isolation Signal (ICCW & NSCCW)	(1)	(1)		(1) When CONTAINMENT INTEGRITY is required.
20. Reactor Building Spray System Logic Channel	NA		NA	
21. Reactor Building Spray 30 psig pressure switches	NA			
22. Pressurizer Temperature Channels		NA		
23. Control Rod Absolute Position	(1)	NA		(1) Check with Relative Position Indicator
24. Control Rod Relative Position	(1)	NA		(1) Check with Absolute Position Indicator
25. Core Flooding Tanks				
a. Pressure Channels Coolant	NA	NA		
b. Level Channels	NA	NA		
26. Pressurizer Level Channels		NA		

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
27. Makeup Tank Instrument Channels:				
a. Level	(1)	NA		(1) When Makeup and Purification System is in operation.
b. Pressure	(1)	NA		
28. Radiation Monitoring Systems*				
a. DELETED				(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that the monitor is functioning.
b. DELETED				
c. DELETED				
d. RM-A2P (RB Atmosphere particulate)	(1)(4)	(4)	(4)	
e. RM-A21 (RB Atmosphere iodine)	(1)(4)	(4)	(4)	(2) DELETED
f. RM-A2G (RB Atmosphere gas)	(1)(4)	(4)	(4)	(3) DELETED
				(4) RM-A2 operability requirements are given in T.S. 3.1.6.8
29. High and Low Pressure Injection Systems: Flow Channels	N/A	N/A		

* Includes only monitors indicated under this item. Other T.S. required radiation monitors are included in specifications 3.5.5.2, 4.1.3, Table 3.5-1 item C.3.f, and Table 4.1-1 item 19e.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
30. Borated Water Storage Tank Level Indicator		NA		
31. DELETED				
32. DELETED				
33. Containment Temperature	NA	NA		
34. Incore Neutron Detectors	(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout when reactor power is greater than 15%.
35. Emergency Plant Radiation Instruments	(1)	NA		(1) Battery Check.
36. (DELETED)				
37. Reactor Building Sump Level	NA	NA		

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
38. OTSG Full Range Level		NA		
39. Turbine Overspeed Trip	NA		NA	
40. Deleted				
41. Deleted				
42. Diesel Generator Protective Relaying	NA	NA		
43. 4 KV ES Bus Undervoltage Relays (Diesel Start)				
a. Degraded Grid	NA	(1)		(1) Relay operation will be checked by local test pushbuttons.
b. Loss of Voltage	NA	(1)		(1) Relay operation will be checked by local test pushbuttons.
44. Reactor Coolant Pressure DH Valve Interlock Bistable	(1)			(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F.
45. Loss of Feedwater Reactor Trip	(1)	(1)		(1) When reactor power exceeds 7% power.
46. Turbine Trip/Reactor Trip	(1)	(1)		(1) When reactor power exceeds 45% power.
47. a. Pressurizer Code Safety Valve and PORV Tailpipe Flow Monitors	(1)	NA		(1) When T_{ave} is greater than 525°F.
b. PORV – Acoustic/Flow	NA	(1)		(1) When T_{ave} is greater than 525°F.
48. PORV Setpoints	NA	(1)		(1) Per Specification 3.1.12 excluding valve operation.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK(c)	TEST(c)	CALIBRATE(c)	REMARKS
49. Saturation Margin Monitor	(1)	(1)		(1) When T_{ave} is greater than 525°F.
50. Emergency Feedwater Flow Instrumentation	NA	(1)		(1) When T_{ave} is greater than 250°F.
51. Heat Sink Protection System				
a. EFW Auto Initiation Instrument Channels				(1) Includes logic test only.
1. Loss of both Feedwater Pumps	NA	(1)		
2. Loss of All RC Pumps	NA	(1)		
3. Reactor Building Pressure	NA			
4. OTSG Low Level				
b. MFW Isolation OTSG Low Pressure	NA			
c. EFW Control Valve Control System				
1. OTSG Level Loops				
2. Controllers		NA		
d. HSPS Train Actuation Logic	NA	(1)		
52. Backup Incore Thermocouple Display	(1)	NA		(1) When T_{ave} is greater than 250°F.
53. Deleted				
54. Reactor Vessel Water Level	NA	NA		

Notes

(a) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. Enter condition into Corrective Action Program.

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

(c) Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Note 1
2. Control Rod Movement	Movement of each rod	Note 1, when reactor is critical
3. Pressurizer Safety Valves	Setpoint	In accordance with the Inservice Testing Program
4. Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5. Refueling System Interlocks	Functional	Start of each refueling period
6. (Deleted)	--	--
7. Reactor Coolant System Leakage	Evaluate	Note 1, when reactor coolant system temperature is greater than 525 degrees F (Not applicable to primary-to-secondary leakage.)
8. (Deleted)	--	--
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation - Visual inspection of Intake Pump House Floor	Note 1
	(b) Silt Accumulation Measurement of Pump House Flow	Note 1
11. Pressurizer Block Valve (RC-V2)	Functional*	Note 1
12. Primary to Secondary Leakage	Evaluate	Note 1 (Note: Not required to be performed until 12 hours after establishment of steady state operation.)

* Function shall be demonstrated by operating the valve through one complete cycle of full travel.

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Verify reactor coolant DOSE EQUIVALENT Xe-133 specific activity is less than or equal to 797 microcuries/gram.	i) Note 1 (during all plant conditions except REFUELING SHUTDOWN and COLD SHUTDOWN). ii) One Sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a one hour period during all plant conditions except REFUELING SHUTDOWN and COLD SHUTDOWN.
	b. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	i) Note 1 (during power operations). ii) One Sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a one hour period during all plant conditions except REFUELING SHUTDOWN and COLD SHUTDOWN. iii) # Once per 4 hours, whenever the specific activity exceeds 0.35 μCi/gram DOSE EQUIVALENT I-131 during all plant conditions except REFUELING SHUTDOWN and COLD SHUTDOWN.
	c. Deleted	
	d. Chemistry (Cl, F and O2)	Note 1 (when Tavg is greater than 200°F).
	e. Boron concentration	Note 1
	f. Tritium Radioactivity	Note 1
2. Borated Water Storage Tank Water Sample	Boron concentration	Note 1 and after each makeup when reactor coolant system pressure is greater than 300 psig or Tavg is greater than 200°F.
3. Core Flooding Tank Water Sample	Boron concentration	Note 1 and after each makeup when RCS pressure is greater than 700 psig.

Item	Check	Frequency
4. Spent Fuel Pool Water Sample	Boron Concentration greater than or equal to 600 ppmb	Note 1
5. Secondary Coolant	Isotopic analysis for DOSE EQUIVALENT I-131 concentration	Note 1 (when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F.
6. Deleted		
7. Deleted		
8. Deleted		
9. Deleted		
10. Deleted		
11. Deleted		
12. Deleted		

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

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** Deleted

*** Deleted

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

TABLE 4.1-4

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	CHECK(a)	TEST(a)	CALIBRATE(a)	REMARKS
1	Noble Gas Effluent				
	a. Condenser Vacuum Pump Exhaust (RM-A5-Hi)				(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that this monitor is functioning.
	b. Condenser Vacuum Pump Exhaust (RM-G25)	(1)			
	c. Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)				
	d. Reactor Building Purge Exhaust (RM-A9-Hi)				
	e. Reactor Building Purge Exhaust (RM-G24)	(1)			
	f. Main Steam Lines Radiation (RM-G26/RM-G27)	(1)			
2.	Containment High Range Radiation (RM-G22/G23)				
3.	Containment Pressure		N/A		
4.	Containment Water Level		N/A		
5.	DELETED				
6.	Wide Range Neutron Flux		N/A		

TABLE 4.1-4 (Continued)

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	CHECK(a)	TEST(a)	CALIBRATE(a)	REMARKS
7.	Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)		N/A		
8.	Reactor Coolant System Hot Leg (TE-958, 960; TI-958A, 960A)		N/A		
9.	Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)		N/A		
10.	Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)		N/A		
11.	Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)		N/A		

(a) Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

TABLE 4.1-5
SYSTEM SURVEILLANCE REQUIREMENTS

Item	Test	Frequency
1. Core Flood Tank	a. Verify two core flood tanks each contain 940 ± 30 ft^3 borated water.	Note 1
	b. Verify that two core flood tanks each contain 600 ± 25 psig.	Note 1
	c. Verify CF-V-1A&B are fully open.	Note 1
	d. Verify power is removed from CF-V-1A&B and CF-V-3A&B valve operators	Note 1
2. Reactor Building Emergency Sump pH Control System	a. Verify the TSP baskets contain ≥ 18,815 lbs and ≤ 28,840 lbs of TSP.	Note 1
	b. Verify that a sample from the TSP baskets provides adequate pH adjustment of borated water.	Note 1

Note 1: Surveillance Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

4.4 REACTOR BUILDING

4.4.1 CONTAINMENT LEAKAGE TESTS

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Testing (ILRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program at test frequencies established in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2 Local Leakage Rate Testing (LLRT) shall be conducted in accordance with the Reactor Building Leakage RateTesting Program. LLRT shall be performed at a pressure not less than peak accident pressure P_{ac} with the exception that the airlock door seal tests shall normally be performed at 10 psig and the periodic containment airlock tests shall be performed at a pressure not less than P_{ac} . LLRT frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Operability of the personnel and emergency air lock door interlocks and the associated control room annunciator circuits shall be determined at the frequency specified in the Surveillance Frequency Control Program. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable, except as provided in Technical Specification Section 3.8.6.

Bases (1)

The Reactor Building is designed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design internal pressure of 55 psig with a coincident temperature of 281°F at accident conditions. The peak calculated Reactor Building pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig. The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}. Containment Isolation Valves are addressed in the UFSAR (Reference 2).

4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM & REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Loading Sequence

Applicability: Applies to periodic testing requirements for safety actuation systems.

Objective: To verify that the emergency loading sequence and automatic power transfer is operable.

Specifications:

4.5.1.1 Sequence and Power Transfer Test

a. At the frequency specified in the Surveillance Frequency Control Program, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable. |

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power.

-M. U. Pump
-D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
-R. B. Cooling Pump
-R. B. Ventilators
-D. H. Closed Cycle Cooling Pump
-N. S. Closed Cycle Cooling Pump
-D. H. River Cooling Pump
-N. S. River Cooling Pump
-D. H. and N. S. Pump Area Cooling Fan
-Screen House Area Cooling Fan
-Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psig Pressure Test Signal.)
-Motor Driven Emergency Feedwater Pump

c. Following successful transfer to the emergency diesel, the diesel generator breaker will be opened to simulate trip of the generator then re-closed to verify block load on the reclosure.

4.5.1.2 Sequence Test

a. At the frequency specified in the Surveillance Frequency Control Program, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power. |

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel.

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification

4.5.2.1 High Pressure Injection

a. At the frequency specified in the Surveillance Frequency Control Program and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

b. The test will be considered satisfactory if the valves (MU-V-14A/B & 16A/B/C/D) have completed their travel and the make-up pumps are running as evidenced by system flow. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A/B/C/D shall be conducted only under either of the following conditions:

1) Indicated RCS temperature shall be greater than 329°F.
2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. At the frequency specified in the Surveillance Frequency Control Program and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps listed in 4.5.1.1b have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.

c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance which affects the position indicator.

4.5.2.3 Core Flooding

a. At the frequency specified in the Surveillance Frequency Control Program, a system test shall be conducted to demonstrate proper operation of the system. Verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flooding tank level verifies that all valves have opened.

4.5.2.4 Component Tests

a. At the frequency specified in the Surveillance Frequency Control Program, the components required for emergency core cooling will be tested.

b. The test will be considered satisfactory if the pumps and fans have been successfully started and the valves have completed their travel as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, verification of pressure/flow, or control board indicating lights initiated by separate limit switch contacts.

Bases

The emergency core cooling systems (Reference 1) are the principal reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

The minimum acceptable HPI/LPI flow assures proper flow and flow split between injection legs.

With the reactor shutdown, the valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

Reference

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

4.5.3 REACTOR BUILDING COOLING AND ISOLATION SYSTEM

Applicability

Applies to testing of the reactor building cooling and isolation systems.

Objective

To verify that the reactor building cooling systems are operable Specification

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At the frequency specified in the Surveillance Frequency Control Program and simultaneously with the test of the emergency loading sequence, a Reactor Building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.

   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during the component test of the R. B. Cooling and Isolation System.

   The test will be considered satisfactory if the spray pumps have been successfully started.

2. Compressed air will be introduced into the spray headers to verify each spray nozzle is unobstructed at the frequency specified in the Surveillance Frequency Control Program.

b. Reactor Building Cooling and Isolation Systems

1. At the frequency specified in the Surveillance Frequency Control Program, a system test shall be conducted to demonstrate proper operation of the system.

2. The test will be considered satisfactory if measured system flow is greater than accident design flow rate.

4.5.3.2 Component Tests

a. At the frequency specified in the Surveillance Frequency Control Program, the components required for Reactor Building Cooling and Isolation will be tested.

b. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, local verification, verification of pressure/flow, or control board component operating lights initiated by separate limit switch contacts.

Bases

The Reactor Building Cooling and Isolation Systems and Reactor Building Spray System are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure (References 1 and 2).

The delivery capability of one Reactor Building Spray Pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump.

With the pumps shut down and the Borated Water Storage Tank outlet valve closed, the Reactor Building spray injection valves can each be opened and closed by the operator action. With the Reactor Building spray inlet valves closed, low pressure air can be blown through the test connections of the Reactor Building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves and instrumentation of the Reactor Building Cooling System are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment.

The Reactor Building fans are normally operating periodically, constituting the test that these fans are operable.

Reference

(1) UFSAR, Section 6.2 - "Reactor Building Spray System"
(2) UFSAR, Section 6.3 - "Reactor Building Emergency Cooling System"

4.5.4 ENGINEERED SAFEGUARDS FEATURE (ESF) SYSTEMS LEAKAGE

Applicability
Applies to those portions of the Decay Heat, Building Spray, and Make-Up Systems, which are required to contain post accident sump recirculation fluid, when these systems are required to be operable in accordance with Technical Specification 3.3.

Objective
To maintain a low leakage rate from the ESF systems in order to prevent significant off-site exposures and dose consequences.

Specification
4.5.4.1 The total maximum allowable leakage into the Auxiliary Building from the applicable portions of the Decay Heat, Building Spray and Make-Up System components as measured during tests in Specification 4.5.4.2 shall not exceed 15 gallons per hour.

4.5.4.2 At the frequency specified in the Surveillance Frequency Control Program the following tests of the applicable portions of the Decay Heat Removal, Building Spray and Make-Up Systems shall be conducted to determine leakage:

a. The applicable portion of the Decay Heat Removal System that is outside containment shall be leak tested with the Decay Heat pump operating, except as specified in “b”.

b. Piping from the Reactor Building Sump to the Building Spray pump and Decay Heat Removal System pump suction isolation valves shall be pressure tested at no less than 55 psig.

c. The applicable portion of the Building Spray system that is outside containment shall be leak tested with the Building Spray pumps operating and BS-V-1A/B closed, except as specified in “b” above.

d. The applicable portion of the Make-Up system on the suction side of the Make-Up pumps shall be leak tested with a Decay Heat pump operating and DH-V-7A/B open.

e. The applicable portion of the Make-Up system from the Make-Up pumps to the containment boundary valves (MU-V-16A/D, 18, and 20) shall be leak tested with a Make-Up pump operating.

f. Visual inspection shall be made for leakage from components of these systems. Leakage shall be measured by collection and weighing or by another equivalent method.

Bases
The leakage rate limit of 15 gph (measured in standard room temperature gallons) for the accident recirculation portions of the Decay Heat Removal (DHR), Building Spray (BS), and Make-Up (MU) systems is based on ensuring that potential leakage after a loss-of-coolant accident will not result in off-site dose consequences in excess of those calculated to comply with the 10 CFR 50.67 limits (Reference 1 and 2). The test methods prescribed in 4.5.4.2 above for the applicable portions of the DH, BS and MU systems ensure that the testing results account for the highest pressure within that system during the sump recirculation phase of a design basis accident.

References
(1) UFSAR, Section 6.4.4 - "Design Basis Leakage"

(2) UFSAR, Section 14.2.2.5(d) - "Effects of Engineered Safeguards Leakage During Maximum Hypothetical Accident"

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability: Applies to periodic testing and surveillance requirement of the emergency power system.

Objective: To verify that the emergency power system will respond promptly and properly when required.

Specification:

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiate start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the name-plate rating (3000 kw). This test will be conducted at the frequency specified in the Surveillance Frequency Control Program on each diesel generator. Normal plant operation will not be effected. |

b. Automatically start and loading the emergency diesel generator in accordance with Specification 4.5.1.1.b/c including the following. This test will be conducted at the frequency specified in the Surveillance Frequency Control Program on each diesel generator. |

   (1) Verify that the diesel generator starts from ambient condition upon receipt of the ES signal and is ready to load in ≤ 10 seconds.

   (2) Verify that the diesel block loads upon simulated loss of offsite power in ≤ 30 seconds.

   (3) The diesel operates with the permanently connected and auto connected load for ≥ 5 minutes.

   (4) The diesel engine does not trip when the generator breaker is opened while carrying emergency loads.

   (5) The diesel generator block loads and operates for ≥ 5 minutes upon reclosure of the diesel generator breaker.

c. Deleted.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded:

   (1) at the frequency specified in the Surveillance Frequency Control Program |
   (2) once within 24 hours after a battery discharge <105 V
   (3) once within 24 hours after a battery overcharge >150 V
   (4) If any cell parameters are not met, measure and record the parameters on each connected cell every 7 days thereafter until all battery parameters are met.

b. The voltage and specific gravity of a pilot cell will be measured and recorded at the frequency specified in the Surveillance Frequency Control Program. If any pilot cell parameters are not met, perform surveillance 4.6.2.a on each connected cell within 24 hours and every 7 days thereafter until all battery parameters are met. |

c. Each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

d. The battery will be subjected to a load test at the frequency specified in the Surveillance Frequency Control Program.

(1) Verify battery capacity exceeds that required to meet design loads.

(2) Any battery which is demonstrated to have less than 85% of manufacturers ratings during a capacity discharge test shall be replaced during the subsequent refueling outage.

4.6.3 Pressurizer Heaters

a. The following tests shall be conducted at the frequency specified in the Surveillance Frequency Control Program:

(1) Pressurizer heater groups 8 and 9 shall be transferred from the normal power bus to the emergency power bus and energized. Upon completion of this test, the heaters shall be returned to their normal power bus.

(2) Demonstrate that the pressurizer heaters breaker on the emergency bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

(3) Verify that following input of the Engineered Safeguards Signal, the circuit breakers, supplying power to the manually transferred loads for pressurizer heater groups 8 and 9, have been tripped.

Bases

The tests specified are designed to demonstrate that one diesel generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an engineered safeguards Actuation Signal. The intent of the periodic tests is to demonstrate the diesel capability to carry design basis accident (LOOP/LOCA) load. The test should not exceed the diesel 2000-hr. rating of 3000 kW. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential overload condition. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the station battery is extremely unlikely. The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

The PORV has a remotely operated block valve to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valve and the block valve is supplied from an ESF power source to ensure the ability to seal this possible RCS leakage path.

The requirement that a minimum of 107 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation.

4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for DHR are capable of performing their design function.

Specification

4.9.1 Reactor Coolant System (RCS) Temperature greater than 250 degrees F.

4.9.1.1 Verify each Emergency Feedwater (EFW) Pump is tested in accordance with the requirements and acceptance criteria of the Inservice Test Program.

Note: This surveillance is not required to be performed for the turbine-driven EFW Pump (EF-P-1) until 24 hours after exceeding 750 psig.

4.9.1.2 DELETED

4.9.1.3 At the frequency specified in the Surveillance Frequency Control Pogram, each EFW System flowpath valve from both Condensate Storage Tanks (CSTs) to the OTSGs via the motor-driven pumps and the turbine-driven pump shall be verified to be in the required status.

4.9.1.4 At the frequency specified in the Surveillance Frequency Control Program:

a) Verify that each EFW Pump starts automatically upon receipt of an EFW test signal.

b) Verify that each EFW control valve responds upon receipt of an EFW test signal.

c) Verify that each EFW control valve responds in manual control from the control room and remote shutdown panel.

4.9.1.5 Prior to STARTUP, following a REFUELING SHUTDOWN or a COLD SHUTDOWN greater than 30 days, conduct a test to demonstrate that the motor driven EFW Pumps can pump water from the CSTs to the Steam Generators.

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4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY-PERIODIC TESTING (Continued)

4.9.1.6 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 RCS Temperature less than or equal to 250 degrees F.*

4.9.2.1 At the frequency specified in the Surveillance Frequency Control Program, verify operability of the means for DHR required by Specification 3.4.2 by observation of console status indication.

* These requirements supplement the requirements of Specifications 4.5.2.2 and 4.5.4.

Bases

The ASME Code specifies requirements and acceptance standards for the testing of nuclear safety related pumps. The EFW Pump test frequency specified by the ASME Code will be sufficient to verify that the turbine-driven and both motor-driven EFW Pumps are operable. Compliance with the normal acceptance criteria assures that the EFW Pumps are operating as expected. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Deferral of the requirement to perform IST on the turbine-driven EFW Pump is necessary to assure sufficient OTSG pressure to perform the test using Main Steam.

Periodic verification of the operability of the required means for DHR ensures that sufficient DHR capability will be maintained.

4.11 REACTOR COOLANT SYSTEM VENTS

Applicability

Applies to Reactor Coolant System Vents.

Objective

To ensure that Reactor Coolant System vents are able to perform their design function.

Specification

4.11.1 Each reactor coolant system vent path shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by cycling each power operated valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

BASES

Tests specified above are necessary to ensure that the individual Reactor Coolant System Vents will perform their functions. It is not advisable to perform these tests during Plant Power Operation, or when there is significant pressure in the Reactor Coolant System. Tests are, therefore, to be performed during either Cold Shutdown or Refueling.

4.12 AIR TREATMENT SYSTEM

4.12.1 EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the emergency control room air treatment system and associated components.

Objective

To verify that this system and associated components will be able to perform its design functions.

Specification

4.12.1.1 At the frequency specified in the Surveillance Frequency Control Program, the pressure drop across the combined HEPA filters and charcoal adsorber banks of AH-F3A and 3B shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).

4.12.1.2 a. The tests and sample analysis required by Specification 3.15.1.2 shall be performed initially and at the frequency specified in the Surveillance Frequency Control Program for standby service or after every 720 hours of system operation and following significant painting, steam, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing which could affect the HEPA filter bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing which could effect the charcoal adsorber bank bypass leakage.

d. Each AH-E18A and B (AH-F3A and B) fan/filter circuit shall be operating at least 10 hours at the frequency specified in the Surveillance Frequency Control Program.

4.12.1.3 At the frequency specified in the Surveillance Frequency Control Program, automatic initiation of the required Control Building dampers for isolation and recirculation shall be demonstrated as operable.

4.12.1.4 An air distribution test shall be performed on the HEPA filter bank initially, and after any maintenance or testing that could affect the air distribution within the system. The air distribution across the HEPA filter bank shall be uniform within ±20%. The test shall be performed at 40,000 cfm (±10%) flow rate.

4.12.1.5 Control Room Envelope unfiltered air inleakage testing shall be performed in accordance with the Control Room Envelope Habitability Program.

4.12.4 FUEL HANDLING BUILDING ESF AIR TREATMENT SYSTEM

Applicability

Applies to Fuel Handling Building (FHB) ESF Air Treatment System and associated components.

Objective

To verify that this system and associated components will be able to perform its design functions.

Specification

4.12.4.1 Each refueling interval prior to movement of irradiated fuel:

a. The pressure drop across the entire filtration unit shall be demonstrated to be less than 7.0 inches of water at 6,000 cfm flow rate (±10%).

b. The tests and sample analysis required by Specification 3.15.4.2 shall be performed.

4.12.4.2 Testing necessary to demonstrate operability shall be performed as follows:

a. The tests and sample analysis required by Specification 3.15.4.2 shall be performed following significant painting, steam, fire, or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank, and after any structural maintenance on the system housing that could affect the HEPA filter bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank, and after any structural maintenance on the system housing that could affect charcoal adsorber bank bypass leakage.

4.12.4.3 Each filter train shall be operated at least 10 hours at the frequency specified in the Surveillance Frequency Control Program.

4.12.4.4 An air flow distribution test shall be performed on the HEPA filter bank initially and after any maintenance or testing that could affect the air flow distribution within the system. The distribution across the HEPA filter bank shall be uniform within ±20%. The test shall be performed at 6,000 cfm ± 10% flow rate.

4.16 REACTOR INTERNALS VENT VALVES SURVEILLANCE

Applicability

Applies to Reactor Internals Vent Valves.

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Specification

	Item	Test	Frequency
4.16.1	Reactor Internals Vent Valves	Demonstrate Operability By: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs. (applied vertically upward).	At the frequency specified in the Surveillance Frequency Control Program

Bases

Verifying vent valve freedom of movement insures that coolant flow does not bypass the core through reactor internals vent valves during operation and therefore insures the conservatism of Core Protection Safety limits as delineated in Figures 2.1-1 and 2.1-3, and the flux/flow trip setpoint.

4.20 REACTOR BUILDING AIR TEMPERATURE

Applicability

This specification applies to the average air temperature of the primary containment during power operations.

Objective

To assure that the temperatures used in the safety analysis of the reactor building are not exceeded.

Specification

4.20.1 When the reactor is critical, the reactor building temperature will be checked at the frequency specified in the Surveillance Frequency Control Program. If any detector exceeds 130°F (120°F below elevation 320) the arithmetic average will be computed to assure compliance with Specification 3.17.1.

6.21 Surveillance Frequency Control Program

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

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

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

c. The provisions of Definition 1.25 and Surveillance Requirement 4.0.2 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-50

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated March 24, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100840205), as supplemented by letters dated July 29, 2010 (ADAMS Accession No. ML102110459), and September 27, 2010 (ADAMS Accession No. ML102700481), Exelon Generation Company, (Exelon, or the licensee) requested changes to the technical specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on May 18, 2010 (75 FR 27829).

The proposed changes would revise the TMI-1 Technical Specifications (TSs) to relocate certain surveillance frequencies to a licensee-controlled program through the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" (ADAMS Accession No. ML071360456). The changes are consistent with U.S. Nuclear Regulatory Commission (NRC, or Commission)-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STSs) change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force Initiative 5b," Revision 3.

When implemented, TSTF-425 relocates most periodic frequencies of TS surveillances to a licensee-controlled program, the Surveillance Frequency Control Program (SFCP), and provides requirements for the new program in the Administrative Controls section of the TS. All surveillance frequencies can be relocated except:

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

A new program is added to the Administrative Controls of TS Section 6 as Specification 6.21. The new program is called the SFCP and describes the requirements for the program to control changes to the relocated surveillance frequencies. The proposed changes to the Administrative Controls of the TS to incorporate the SFCP include a specific reference to NEI 04-10 as the basis for making any changes to the surveillance frequencies once they are relocated out of the TS.

In a letter dated September 19, 2007, the NRC staff approved NEI 04-10, Revision 1, (ADAMS Accession No. ML072570267), as acceptable for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, and the safety evaluation providing the basis for NRC acceptance of NEI 04-10.

2.0 REGULATORY EVALUATION

In its "Final Policy Statement: Technical Specifications for Nuclear Power Plants," published in the FR (58 FR 39132-39139, July 22, 1993) the NRC addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment or PRA) in STSs. The Commission established four criteria to identify those constraints on design and operation of nuclear power plants that are derived from the plant safety analysis report or PSA information, that are required to remain in a plants TSs. These were as follows:

> *Criterion 1*
> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> *Criterion 2*
> A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> *Criterion 3*
> A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> *Criterion 4*
> A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

(58 FR 39136-39137).

In discussing the use of PSA in Nuclear Power Plant Technical Specifications, the Commission wrote, in part:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results

> of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed. ....
>
> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants", 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "*** probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made *** about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process of determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety." ....
>
> 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.

(58 FR 39135).

Approximately 2 years later, the NRC provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," published in the FR (60 FR 42622-42629, August 16, 1995). The Commission, in discussing the deterministic and probabilistic approaches to regulation, and the Commission's extension and enhancement of traditional regulation, wrote in part (60 FR 42627):

> 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.

The Commission then provided its new policy, stating:

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

through more efficient use of agency resources; and through a reduction in unnecessary burdens on licensees.

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 [Title 10 of the *Code of Federal Regulations*] [(]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.

(60 FR 42628-42629).

In 10 CFR 50.36, 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; (3) surveillance requirements; (4) design features; and (5) administrative controls.

As stated in 10 CFR 50.36(c)(3), "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." These categories will remain in the TMI-1 TSs. The new TS, SFCP, provides the necessary administrative controls to require that surveillances relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, including qualitative considerations; results of risk analyses, sensitivity studies and any bounding analyses; and recommended monitoring of structures, systems, and components (SSCs). It is required that these changes be documented. Furthermore, changes to frequencies are subject to regulatory review and oversight of the SFCP implementation through the rigorous NRC review of safety-related SSC performance provided by the reactor oversight program.

Licensees are required by TSs to perform surveillance test, calibration, or inspection on specific safety-related system equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies, currently identified in TSs, are based primarily upon deterministic methods such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved methodologies identified in NEI 04-10 provides a way to establish risk-informed surveillance frequencies that complement the deterministic approach and support the NRC's traditional defense-in-depth philosophy.

The licensee's SFCP ensures that surveillance requirements specified in the TSs are performed at intervals sufficient to assure the above regulatory requirements are met. Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and 10 CFR Part 50 Appendix B, Quality Assurance Criteria, Section XVI, Corrective Action, require licensee monitoring of surveillance test failures and implementation of corrective actions to address such failures. One of these actions may be to consider increasing the frequency at which a surveillance test is performed. In addition, the SFCP implementation guidance in NEI 04-10 requires monitoring the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs. These requirements, and the monitoring required by NEI 04-10, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified and appropriate corrective actions taken.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1 (ADAMS Accession No. ML023240437), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML003740176), describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in allowed outage times. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed completion time (CT) TS change, as discussed below.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability and incremental conditional large early release probability. Tier 1 also addresses PRA quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the present TS change in light of past related applications or additional applications under review are also considered along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when the proposed CT is implemented.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule [10 CFR 50.65(a)(4)], which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 (ADAMS Accession No. ML070240001), describes an acceptable approach, for submittals made prior to April 2010, for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light water-reactors.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425 for TMI-1 provides for administrative relocation of applicable surveillance frequencies, and provides for the addition of the SFCP to the administrative controls of TSs. TSTF-425 also requires the application of NEI 04-10 for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes proposed in TSTF-425 included documentation regarding the PRA technical adequacy consistent with the requirements of RG 1.200, Revision 1. In accordance with NEI 04-10, PRA methods are used, in combination with plant performance data and other considerations, to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is in accordance with guidance provided in RG 1.174 and RG 1.177 in support of changes to surveillance test intervals.

3.1 RG 1.177 Five Key Safety Principles

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

3.1.1 The Proposed Change Meets Current Regulations

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

Exelon has proposed to add Section 6.21 of the TSs which requires any changes to the SR frequencies to be made in accordance with NEI 04-10, Revision 1. By letter dated September 19, 2007, the NRC staff found that NEI 04-10, Revision 1, met NRC regulations, specifically 10 CFR 50.36(c), and was an acceptable control program for this type of application. Thus, this proposed change meets the first key safety principle of RG 1.177, by complying with current regulations.

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

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

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not impacted.
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, is maintained.

The proposed TS Section 6.21 would require the application of NEI 04-10 for any changes to surveillance frequencies within the SFCP. NEI 04-10 uses both the CDF and the LERF metrics to evaluate the impact of proposed changes to surveillance frequencies. The guidance of RG 1.174 and RG 1.177 for changes to CDF and LERF is achieved by evaluation using a comprehensive risk analysis, which assesses the impact of proposed changes including contributions from human errors and common cause failures. Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to an increased likelihood of common cause failures. Both the quantitative risk analysis and the qualitative considerations assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177.

3.1.3 The Proposed Change Maintains Sufficient Safety Margins

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

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

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

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

RG 1.177 provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies. This requires the identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. The proposed TS 6.21 would require the application of NEI 04-10 in the SFCP. As previously discussed, NEI 04-10 has previously been found by the NRC staff to satisfy the intent of the RG 1.177 requirements for evaluating the change in risk, and for assuring that such changes are small.

3.1.4.1 Quality of the PRA

The quality of the TMI-1 PRA is compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the quality of the PRA.

The licensee used RG 1.200, Revision 1, to address the technical adequacy of the TMI-1 PRA. RG 1.200 is NRC's developed regulatory guidance, which endorses with comments and qualifications the use of the American Society of Mechanical Engineers ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to ASME RA-S-2002, 'Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,'" dated December 30, 2005; NEI 00–02, "PRA Peer Review Process Guidelines," dated March 20, 2000; and NEI 05–04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard," dated January 2005. The licensee has performed an assessment of the PRA models used to support the SFCP against the requirements of RG 1.200 to assure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability Category II (CC-II) was applied as the standard, and any identified deficiencies to those requirements were assessed further to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies, where appropriate.

The licensee reported that several assessments of the technical capability of its PRA have been made, as follows:

- An independent PRA peer review by the B&W Owners Group in 2000, following the Industry PRA peer review process and including an assessment of the PRA model maintenance and update process.

- A limited scope gap assessment in 2005 to support the Mitigating Systems Performance Indicator (MSPI) implementation, including an additional evaluation by the B&W Owners Group PRA via a cross-comparison study to support implementation of the MSPI process.

- A RG 1.200 peer review in 2008 against the 2005 (ASME RA-Sb-2005) and 2007 (ASME RA-Sc-2007) versions of the ASME PRA Standard, excluding the DA [Data] and IF [Internal Flooding] elements.

The licensee completed the most recent PRA model update (TM1080 version) in 2009, including changes to address most of the identified gaps from the 2008 peer review, as well as to address other open items. The licensee summarized 36 open peer review findings in Table 2-1 of its License Amendment Request (LAR) dated March 24, 2010, Attachment 2, excluding the unreviewed DA and IF elements. Both of these elements have been upgraded since the original 2000 peer review, with the results of a self-assessment against the 2005 and 2007 versions of the ASME PRA Standard, provided in Table 2-2 of its LAR dated March 24, 2010, Attachment 2, in terms of gaps with respect to CC-II (16 were identified, 13 for DA and three for IF).

The licensee plans to review all remaining gaps for consideration in the next periodic PRA model update, judging them to currently have a low impact on the PRA model and its ability to support a full range of PRA applications. The remaining gaps are documented in a database for tracking their potential impacts on applications where appropriate. Each item will be reviewed as part of each surveillance test interval (STI) change assessment, including an assessment of any impact on the application. If a non-trivial impact is expected, then this will require further treatment, such as the performance of additional sensitivity studies or model changes.

The staff reviewed the licensee's assessment of the TMI-1 PRA and the remaining open peer review findings, including those related to elements DA and IF. Results from the review of the two tables are discussed separately below.

3.1.4.1.1 PRA Quality – Open Peer Review Findings

Over half of the open findings (LAR, Attachment 2, Table 2-1) were deemed to be "documentation issue[s] not affecting the adequacy of the PRA model." The remainder of the open findings were dispositioned to be "addressed by sensitivities per NEI 04-10, if applicable to the specific STI evaluation." While the NRC staff agreed with the majority of these dispositions, clarifications were requested for certain findings as detailed in an NRC request for additional information (RAI) dated July 2, 2010 (ADAMS Accession No. ML101680647). Specifically, the licensee was requested to address whether, when taken cumulatively, the effects of the findings could prove significant to the risk evaluation for an STI change. In its July 29, 2010, response, the licensee provided such an assessment. The NRC staff reviewed the licensee's assessment and agrees that, even when taken cumulatively, the effects from these issues/gaps do not significantly impact the results from an STI risk evaluation performed via the NEI 04-10 methodology.

3.1.4.1.2 PRA Quality – Gaps for DA and IF Elements to CC-II

Several of the DA/IF Gaps (LAR, Attachment 2, Table 2-2) were deemed to be "documentation issue[s] not affecting the adequacy of the PRA model," with which the NRC staff agrees. The remaining Gaps (over half) were dispositioned to be "addressed by sensitivities per NEI 04-10, if applicable to the specific STI evaluation." While the staff agreed with some of these dispositions, clarifications for certain dispositions, as detailed in NRC RAI dated July 2, 2010, were requested. Specifically the NRC staff questioned whether certain outliers in the definition of system/component failure groups were excluded from group definitions, or if not, would their exclusion be part of the sensitivity analysis for an STI evaluation.

The licensee responded to this question by letter dated July 29, 2010. Based on its review of this response, the staff finds that, although not excluded in the base PRA model, the licensee will appropriately exclude any outliers when performing the required sensitivity analysis for an STI evaluation, which meets the requirements of NEI 04-10.

Also, regarding the potential for human-induced floods, the NRC staff inquired about valves that could receive new STIs and whether a revised STI could increase the frequency of a flood due to mis-calibration etc. The licensee responded to this question by letter dated July 29, 2010, detailing how this potential would be accounted for. Specifically, the licensee stated that *"...the methodology requires sensitivities for assumptions in the PRA model that may affect the results of the analysis or of any gaps to Capability Category II. This would lead to these issues being appropriately addressed for any valves associated with a surveillance interval change analysis."* Based on this response, the staff is satisfied that if the licensee identifies a valve as a potential flooding source when evaluating an STI, such that increasing the STI could increase the frequency of a flood due to mis-calibration, etc., this potential will be addressed prior to making the STI change.

Based on the licensee's assessment using the 2005 and 2007 versions of the ASME PRA Standard and RG 1.200, the level of PRA quality, combined with the proposed evaluation and disposition of gaps, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 of RG 1.177.

3.1.4.2 Scope of the PRA

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

Exelon evaluated external hazards for TMI-1 via the Individual Plant Examination for External Events (IPEEE) submittal in accordance with NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991. This constituted 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 TMI-1 IPEEE did not screen out seismic or fire hazards, but provided quantitative analyses based on a detailed Seismic PRA and a combination of the Fire Induced Vulnerability Evaluation (FIVE) methodology and Fire PRA.

Subsequently, in 2005, Exelon developed an updated TMI-1 Fire PRA model and updated it in 2007 to incorporate a partial implementation of NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities," EPRI 1011989, September 2005. However, the TMI-1 Fire PRA has not undergone a PRA peer review, and therefore, its use is limited to obtaining additional insights and providing qualitative and bounding quantitative assessments.

The TMI-1 IPEEE also evaluated the following other external hazards, finding each to be a non-significant contributor to plant risk: (1) extreme winds/tornadoes, (2) aircraft crash, (3) transportation accidents, (4) nearby facility accidents, and (5) external floods (based on its location by the Susquehanna River). These assessments are not maintained and would be used for qualitative insights only.

NEI 04-10 allows STI change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. Therefore, in performing the assessments for the other hazard groups, Exelon will perform a qualitative or bounding analysis to provide justification for the acceptability of the proposed test interval change. The fire PRA model will be exercised to obtain quantitative fire risk insights when appropriate, with refinements as needed on a case-by-case basis.

The NRC staff concludes that the licensee's evaluation methodology is sufficient to ensure the scope of the risk contribution of each surveillance frequency change is properly identified for evaluation, and is consistent with Regulatory Position 2.3.2 of RG 1.177.

3.1.4.3 PRA Modeling

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

The licensee will perform quantitative evaluations of the impact of selected testing strategy (i.e., staggered testing or sequential testing) consistent with the guidance of NUREG/CR–6141, "Handbook of Methods for Risk-Based Analysis of Technical Specifications," and NUREG/CR-5497, "Common-Cause Failure Parameter Estimations," as discussed in NEI 04-10.[1]

Thus, through the application of NEI 04-10, the TMI-1 PRA modeling is sufficient to ensure an acceptable evaluation of risk for the proposed changes in surveillance frequency, and is consistent with Regulatory Position 2.3.3 of RG 1.177.

3.1.4.4 Assumptions for Time Related Failure Contributions

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

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

3.1.4.5 Sensitivity and Uncertainty Analyses

NEI 04-10 requires sensitivity studies to assess the impact of uncertainties from key assumptions of the PRA or in the failure probabilities of the affected SSCs, impact to the frequency of initiating events, and impact of any identified deviations from the ASME 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

[1] Note that the TMI-1 TSs do not contain a definition for Staggered Test Basis, thus the deletion of this definition from the TS, as called for in TSTF-425, is not necessary.

may bear on the results of such sensitivity studies. Required monitoring and feedback of SSC performance, once the revised surveillance frequencies are implemented, will also be performed. Thus, through the application of NEI 04-10, the licensee has appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, and its approach is consistent with Regulatory Position 2.3.5 of RG 1.177.

3.1.4.6 Acceptance Guidelines

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

In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact below 1E–5 per year for change to CDF, and below 1E–6 per year for change to LERF, and the total CDF and total LERF must be reasonably shown to be less than 1E–4 per year and 1E–5 per year, respectively. These are consistent with the limits of RG 1.174 for acceptable changes in risk, as referenced by RG 1.177 for changes to surveillance frequencies. The staff further notes that, by adopting the NEI 04-10 methodology for STI TS changes, Exelon includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with negligibly small risk increases (less than 5E–8 per year CDF and 5E–9 per year LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

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

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

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

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

3.2 Addition of Surveillance Frequency Control Program to Administrative Controls

The licensee has included the SFCP and specific requirements in sub-section 6.21, "Surveillance Frequency Control Program," in TS Section 6, "Administrative Controls," as follows:

> This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure that the associated Limiting Conditions for Operation are met.
>
> a. The Surveillance Frequency Control Program shall contain a list of Frequencies of the Surveillance Requirements for which the Frequency is controlled by the program.
>
> b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
>
> c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

The NRC staff finds that the proposed addition to the Administrative Controls section of the TSs adequately identifies the scope of the SFCP and defines the methodology to be used in a revision of SR frequencies.

3.3 Summary and Conclusions

The NRC staff has reviewed the licensee's proposed relocation of some surveillance frequencies to a licensee-controlled document, and its proposal to control changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the Administrative Controls of TS. The SFCP and TS 6.21 references NEI 04-10, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TS to a licensee-controlled document, provided those frequencies are changed in accordance with NEI 04-10, which is specified in the Administrative Controls of the TSs.

The proposed licensee adoption of TSTF-425 and risk-informed methodology 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 the 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.

10 CFR 50.36(c)(3) states that Technical Specifications will include Surveillance Requirements. It further states that, "Surveillance Requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The NRC staff finds that, with the proposed relocation of surveillance frequencies to an owner-controlled document that is administratively controlled in accordance with the TS SFCP, Exelon continues to meet the regulatory requirements of 10 CFR 50.36, and specifically, 10 CFR 50.36(c)(3), Surveillance requirements.

3.4 TS Bases

Because the TMI-1 TS Bases are integrated into the TS, some of the issued pages contain bases revisions associated with the proposed change. The revised wording in the bases is included only for ease of implementation and does not imply NRC staff review or approval of their content.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The state official asked the NRC staff if this change had been done before at any other plant and why this change would be helpful and/or necessary. The NRC staff provided the state official with a partial listing of plants for which this change had been granted and a short summary of the change rationale, based on the background section of TSTF-425. The State official had no further comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes inspection and surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (75 FR 27829). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: R. Gallucci
P. Bamford

Date: January 12, 2011

Please contact me at 301-415-2833 if you have any questions.

Sincerely,

/ra/

Peter J. Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 274 to DPR-50
2. Safety Evaluation

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