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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Response to Request for Additional Information, Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

References:

1. Letter from Pamela B. Cowan, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," dated March 24, 2010.

2. Letter from Peter Bamford, U.S. Nuclear Regulatory Commission, to Michael J. Pacilio, Exelon Nuclear, "Three Mile Island Nuclear Station, Unit 1 - Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Relocation of Surveillance Frequencies to a Licensee-Controlled Program (TAC No. ME3587)," dated September 8, 2010.

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI Unit 1). The proposed amendment would modify TMI Unit 1 TS by relocating selected Surveillance Requirement frequencies to a licensee-controlled program. The NRC reviewed the license amendment request and identified the need for additional information in order to complete their evaluation of the amendment request. On August 17, 2010, draft questions were sent to Exelon to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. In Reference 2, the NRC formally issued the request for additional information (RAI). Attachment 1 to this letter provides a restatement of the questions along with Exelon's responses. Attachment 2 provides revised proposed TS/Bases markups in response to RAI-1.

It should be noted that the revised proposed TS/Bases pages provided in Attachment 2 to this letter are specific to the issue identified in RAI-1, and supersede only the corresponding TS/Bases pages provided in the original submittal (Reference 1). As a result, the pages provided are a subset of the total TS/Bases pages provided in the original submittal and are not meant to be a complete replacement set of proposed TS/Bases pages for the license amendment request. Therefore, the TS/Bases pages provided in the Reference 1 submittal that are not included in Attachment 2 to this letter are still valid and remain part of the requested license amendment.

Exelon has concluded that the information provided in this response does not impact the conclusions provided in the original submittal (Reference 1).

This response to the request for additional information contains no regulatory commitments.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27th day of September 2010.

Respectfully,

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information
Attachment 2: Revised Proposed Technical Specifications/Bases Pages

cc:	Regional Administrator - NRC Region I	w/attachments
	NRC Senior Resident Inspector - TMI Unit 1	"
	NRC Project Manager, NRR - TMI Unit 1	"
	Director, Bureau of Radiation Protection - PA Department of Environmental Resources	"
	Chairman, Board of County Commissioners of Dauphin County	"
	Chairman, Board of Supervisors of Londonderry Township	"

ATTACHMENT 1

License Amendment Request

Three Mile Island Nuclear Station, Unit 1
Docket No. 50-289

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Response to Request for Additional Information

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
APPLICATION FOR TECHNICAL SPECIFICATION CHANGE REGARDING RISK-INFORMED JUSTIFICATION FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE CONTROLLED PROGRAM
(ADOPTION OF TSTF-425, REVISION 3)**

In Reference 1, Exelon Generation Company, LLC (Exelon) submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI Unit 1). The proposed amendment would modify TMI Unit 1 TS by relocating selected Surveillance Requirement frequencies to a licensee-controlled program. The NRC reviewed the license amendment request and identified the need for additional information in order to complete their evaluation of the amendment request. On August 17, 2010, draft questions were sent to Exelon to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. In Reference 2, the NRC formally issued the request for additional information (RAI). The questions are restated below along with Exelon's responses.

RAI-1

Certain of the proposed specifications do not clearly delineate that only the required frequency of surveillance is controlled by the surveillance frequency control program (SFCP). For example proposed TS 3.5.2.4.g states that “Quadrant tilt shall be monitored… in accordance with the surveillance frequency control program…” In order to clearly delineate what the SFCP is authorized to control, this wording should be “Quadrant tilt shall be monitored… *at a frequency* in accordance with the surveillance frequency control program…” or similar wording (italics added). This applies to proposed TSs located on pages 3-34a, 3-35a, 3-59, 4-3, 4-29, 4-39, 4-41, 4-42, 4-43, 4-44, 4-45, 4-46, 4-47, 4-52, 4-52a, 4-54, 4-55, and 4-86 of the LAR. Please provide revised specification pages that incorporate this clarification.

RESPONSE

To make it clear that the Surveillance Frequency Control Program only controls changes to the frequency of surveillances that are included in the program, the proposed wording "in accordance with the Surveillance Frequency Control Program" inserted in the TS and Bases markups has been replaced with the wording "at the frequency specified in the Surveillance Frequency Control Program" as indicated in the revised proposed TS/Bases pages provided in Attachment 2.

RAI-2

Proposed revised TS 3.14.1.1, relating to periodic inspection of the dikes around TMI, is a site specific specification not contained in NUREG-1430 “Standard Technical Specifications Babcock and Wilcox Plants,” and therefore not included as a line item in TSTF-425. As described in the LAR, Attachment 3, page 3-59, this surveillance relates directly to protection against external flooding. Attachment 2 of the LAR, page 21 of 23, states that external flooding was evaluated in the IPEEE [Individual Plant Examination for External Events] and a CDF [core damage frequency] estimated, and that this evaluation has not been maintained. Please describe, for a relocated dike surveillance frequency, how an analysis for a surveillance

frequency change of the dike inspections would incorporate recent site-specific information and use up-to-date databases, if the external flooding analysis is not maintained. Alternatively, please submit a revised application removing the dike surveillance from the scope of this LAR.

RESPONSE

The process for using a Probabilistic Risk Assessment (PRA) to perform an evaluation of a surveillance frequency change is described in the NEI 04-10, Revision 1 methodology invoked by TSTF-425, Revision 3. The methodology requires that the analyst ensure that the PRA is adequate to perform the analysis. If not, a qualitative or bounding analysis can be performed, or the PRA may be changed. The NEI 04-10 process must be followed, as required by the proposed change to TMI TS Section 6.21, if the TS 3.14.1.1 surveillance frequency for performing periodic inspection of the dikes around TMI is to be modified. If it is determined that the IPEEE external flooding analysis is not adequate to quantitatively evaluate a proposed change to the TS 3.14.1.1 surveillance frequency, then modifying the external flooding PRA model if possible to adequately perform a quantitative evaluation would be required, or a qualitative or bounding analysis would be performed. An update to the external flood analysis would typically include consideration of the following:

- more recent information on flood frequencies and flow rates, e.g., the Probable Maximum Flood (PMF),
- use of the current internal events PRA model as a basis for the external flooding PRA, to reflect the as-built and as-operated plant,
- the most recent flood mitigation procedures, equipment, or strategies, and
- current or updated failure rates and probabilities for significant structures, systems and components.

REFERENCES:

1. Letter from Pamela B. Cowan, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)," dated March 24, 2010.

2. Letter from Peter Bamford, U.S. Nuclear Regulatory Commission, to Michael J. Pacilio, Exelon Nuclear, "Three Mile Island Nuclear Station, Unit 1 - Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Relocation of Surveillance Frequencies to a Licensee-Controlled Program (TAC No. ME3587)," dated September 8, 2010.

ATTACHMENT 2

License Amendment Request

Three Mile Island Nuclear Station, Unit 1
Docket No. 50-289

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Revised Proposed Technical Specifications/Bases Pages

3-34a	4-29	4-45	4-54
3-35a	4-39	4-46	4-55
3-59	4-41	4-47	4-55a
4-2a	4-42	4-48	4-59
4-2d	4-43	4-52	4-86
4-3	4-44	4-52a	

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 ~~every 7 days~~**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 ~~on a minimum frequency of once every 12 hours~~**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 least once every six months~~**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”

Bases (Cont'd)

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked **at the frequency specified in the Surveillance Frequency Control Program against a heat balance standard** and calibrated if necessary~~, every shift against a heat balance standard~~. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the ~~intervals of each refueling period~~**frequency specified in the Surveillance Frequency Control Program**.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth **in the Surveillance Frequency Control Program** are considered acceptable.

Testing

On-line testing of reactor protection channels is required ~~semi-annually~~ **at the frequency specified in the Surveillance Frequency Control Program** on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel (Reference 1). **Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.**

~~The rotation schedule for the reactor protection channels is as follows:~~

~~a) Deleted~~
~~b) Semi-annually with one channel being tested every 46 days on a continuous sequential rotation.~~

~~The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested every 46 days. The frequency of every 46 days on a continuous sequential rotation is consistent with the calculations of Reference 2 that indicate the RPS retains a high level of reliability for this interval.~~

Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested **at the frequency specified in the Surveillance Frequency Control Program**~~quarterly with one channel being tested every 23 days on a continuous sequential rotation. Calculations have shown that the frequency of every 23 days maintains a high level of reliability of the Reactor Trip System in Reference 4~~. The trip test checks all logic combinations and is to be performed on a rotational basis.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

Bases (Cont'd)

The TSP is stored in wire mesh baskets placed inside the containment at the 281 ft elevation. Any quantity of TSP between 18,815 lb and 28,840 lb. will result in a pH in the desired range. If it is discovered that the TSP in the containment building is not within limits, action must be taken to restore the TSP to within limits. The Completion Time of 72 hours is allowed for restoring the TSP within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components.

Surveillance Testing

Periodic determination of the mass of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. ~~A Refueling Frequency~~**The surveillance** is required to determine that ≥ 18,815 lbs and ≤ 28,840 lbs are contained in the TSP baskets. This requirement ensures that there is an adequate mass of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.3 and ≤ 8.0. The periodic verification is required ~~every refueling outage~~**at the frequency specified in the Surveillance Frequency Control Program**. ~~Operating experience has shown this Surveillance Frequency to be acceptable due to the margin in the mass of TSP placed in the containment building.~~

Periodic testing is performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. Satisfactory completion of this test assures that the TSP in the baskets is "active." Adequate solubility is verified by submerging a representative sample, taken via a sample thief or similar instrument, of TSP from one of the baskets in containment in un-agitated borated water heated to a temperature representing post-LOCA conditions; the TSP must completely dissolve within a 4 hour period. The test time of 4 hours is to allow time for the dissolved TSP to naturally diffuse through the un-agitated test solution. Agitation of the test solution during the solubility verification is prohibited, since an adequate standard for the agitation intensity (other than no agitation) cannot be specified. The agitation due to flow and turbulence in the containment sump during recirculation would significantly decrease the time required for the TSP to dissolve. Adequate buffering capability is verified by a measured pH of the sample solution, following the solubility verification, between 7.3 and 8.0. The sample is cooled and thoroughly mixed prior to measuring pH. The quantity of the TSP sample, and quantity and boron concentration of the water are chosen to be representative of post-LOCA conditions. ~~A sampling Frequency of every refueling outage is specified. Operating experience has shown this Surveillance Frequency to be acceptable.~~

REFERENCE

(1) UFSAR, Section 7.1.2.3(d) - "Periodic Testing and Reliability"
(2) NRC SER for BAW-10167A, Supplement 1, December 5, 1988.
(3) BAW-10167, May 1986.
(4) BAW-10167A, Supplement 3, February 1998.
(5) EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

TABLE 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHECK**(c)**	TEST**(c)**	CALIBRATE**(c)**	REMARKS
1. Protection Channel Coincidence Logic	NA	~~Q~~	NA	
2. Control Rod Drive Trip Breaker and Regulating Rod Power SCRs	NA	~~Q~~	NA	(1) Includes independent testing of shunt trip and undervoltage trip features.
3. Power Range Amplifier	~~D~~(1)	NA	(2)	(1) When reactor power is greater than 15%. (2) When above 15% reactor power run a heat balance check ~~once per shift~~**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	~~S~~	~~S/A~~	~~M~~(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	~~S~~(1)	~~P S/U~~	NA	(1) When in service.
6. Source Range Channel	~~S~~(1)	~~P S/A~~	NA	(1) When in service.
7. Reactor Coolant Temperature Channel	~~S~~	~~S/A~~	~~F~~	

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 least once per six months~~ **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. ~~During each refueling interval~~**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 intervals not to exceed 3 months~~**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. ~~During each refueling interval~~ **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. ~~During each refueling period~~ **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. ~~During each refueling period~~**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 intervals not to exceed 3 months~~**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 each refueling interval~~ **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 least every ten years~~**at the frequency specified in the Surveillance Frequency Control Program**.

b. Reactor Building Cooling and Isolation Systems

1. ~~During each refueling period~~**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 intervals not to exceed three months~~**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 ravel 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 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 ~~refueling interval~~ tests in Specification 4.5.4.2 shall not exceed 15 gallons per hour.

4.5.4.2 ~~Once each refueling interval~~ **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.

c. 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.

d. 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 ~~every month~~ **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 ~~every refueling interval~~ **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) ~~every 92 days~~**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 ~~weekly~~**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 ~~on a refueling interval basis~~**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 least once each refueling~~ **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 ~~monthly~~ **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 **S**~~s~~urveillance ~~specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails~~**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.

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the axial power shaping rods (APSRs), from the fully withdrawn position to ¾ insertion (104 inches travel) shall not exceed 1.66 seconds at hot reactor coolant full flow conditions or 1.40 seconds for the hot no flow conditions (Reference 1). For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has actuated the 25% withdrawn reference switch during insertion from the fully withdrawn position. The specified trip time is based upon the safety analysis in UFSAR, Chapter 14 and the Accident Parameters as specified therein.

Each control rod drive mechanism shall be exercised by a movement of a minimum of 3% of travel ~~every 92 days~~**at the frequency specified in the Surveillance Frequency Control Program**. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

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 least once per 31 days~~**At the frequency specified in the Surveillance Frequency Control Program**, 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 ~~On a refueling interval basis~~**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.

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 ~~On a daily basis~~**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 ~~quarterly~~ 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.

~~Daily~~ **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 ~~once per refueling interval~~ **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

~~Frequency of t~~**T**ests 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 least every refueling interval~~**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 least once per year~~ **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 ~~every month~~**at the frequency specified in the Surveillance Frequency Control Program**.

4.12.1.3 ~~At least once per refueling interval~~**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.

BASES

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined ~~at least once per refueling cycle~~ **at the frequency specified in the Surveillance Frequency Control Program** to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with approved test procedures. Replacement adsorbent should be qualified according to ASTM D3803-1989. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable all adsorbent in the system shall be replaced. Tests of the HEPA filters with DOP aerosol shall also be performed in accordance with approved test procedures. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Guide 1.52 March 1978.

Operation of the system for 10 hours ~~every month~~ **at the frequency specified in the Surveillance Frequency Control Program** will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

If significant painting, steam, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the Vice President-TMI Unit 1.

Demonstration of the automatic initiation of the recirculation mode of operation is necessary to assure system performance capability. Dampers required for control building isolation and recirculation are specified in UFSAR Sections 7.4.5 and 9.8.1.

Control Room Envelope unfiltered air inleakage testing verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. Air inleakage testing verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Section 3.15.1.5 must be entered. The required actions allow time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 1) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 2). These compensatory measures may also be used as mitigating actions as required by Section 3.15.1.5. Temporary analytical methods may also be used as compensatory measures to

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).	~~Each Refueling Shutdown~~**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 ~~once each twenty-four (24) hours~~ **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.