Michael P. Gallagher
Exelon Nuclear
Vice President
License Renewal and Decommissioning

200 Exelon Way
Kennett Square, PA 19348

610 765 5958 Office
610 765 5658 Fax
www.exeloncorp.com
michaelp.gallagher@exeloncorp.com

10 CFR 50.90

TM-18-079

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

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

Subject: License Amendment Request – Proposed Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition

1. Letter from J. Bradley Fewell (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Certification of Permanent Cessation of Power Operations for Three Mile Island Nuclear Station, Unit 1," dated June 20, 2017 (ML17171A151)
2. Letter from Michael P. Gallagher (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Changes to Technical Specifications Sections 1.0, 'Definitions,' and 6.0, 'Administrative Controls' for Permanently Defueled Condition," dated November 10, 2017 (ML17314A024)
3. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 – Approval of Certified Fuel Handler Training and Retraining Program (CAC NO. MF9960; EPID L-2017-LLL-0013)," dated December 29, 2017 (ML17228A729)

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (Exelon) requests amendments to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The proposed amendment would revise the RFOL and the associated TS to the Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. Exelon is also proposing changes to the Current Licensing Basis (CLB) mitigation strategies for Flood Mitigation and Aircraft Impact Protection in the Air Intake Tunnel.

By letter dated June 20, 2017 (Reference 1), Exelon provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) of Exelon's contingent determination to permanently cease operations at TMI-1 no later than September 30, 2019. Once the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC pursuant to 10 CFR 50.82(a)(1)(i) and (ii), NRC regulations stipulated in 10 CFR 50.82(a)(2) will no longer authorize operation of the reactor or emplacement of fuel into the reactor vessel under the 10 CFR Part 50 license. In support of this condition, the TMI-1 RFOL and associated TS are being proposed for revision to reflect the planned permanent shutdown and defueled condition in accordance with 10 CFR 50.36(c)(6).

The bases for the proposed amendment is that certain license conditions, TS requirements, and CLB may be revised or removed to reflect the permanently defueled condition. In general, the changes propose the elimination of those TS applicable in operating conditions where fuel is placed in the reactor vessel. Changes to other TS limiting conditions for operation, definitions, surveillance requirements, administrative controls, as well as several license conditions are also proposed. The proposed amendment would modify the 10 CFR Part 50 License and the TS to make those changes.

The NRC is currently reviewing proposed changes to the organization, staffing, and training requirements contained in Section 6, "Administrative Controls," of the TMI-1 TS that were submitted November 10, 2017 (Reference 2). The NRC approved the Certified Fuel Handler Training and Retraining Program on December 29, 2017 (Reference 3). The two referenced licensing actions complement and support this proposed license amendment request.

Attachment 1 to this letter provides a detailed description and evaluation of the proposed changes. Attachment 2 contains a markup of the current RFOL and TS pages, including Bases (TS sections that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 2).

The proposed changes have been reviewed and approved by the station's Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92.

Exelon requests review and approval of this proposed amendment by July 25, 2019. Exelon requests that the approved amendment become effective following the submittal of the required 10 CFR 50.82(a)(1)(ii) certification that TMI-1 has been permanently defueled. Implementation is proposed to be completed within 30 days of the effective date. As discussed in Attachment 1, a 60-day decay period is needed to meet the dose limits for the Fuel Handling Accident. In order to implement the PDTS within 30-days following permanent defueling, Exelon is requesting a license condition, as proposed in Attachment 1, that restricts fuel handling during that 60-day time period.

There are no regulatory commitments contained within this submittal.

In accordance with 10 CFR 50.91 "Notice for public comment; State consultation" paragraph (b), Exelon is notifying the State of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions concerning this submittal, please contact Paul Bonnett at (610) 765-5264.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of July 2018.

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President, License Renewal & Decommissioning
Exelon Generation Company, LLC

Attachments: 1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specifications Pages

cc: w/Attachments

NRC Regional Administrator, Region I
NRC Senior Resident Inspector – Three Mile Island Nuclear Station – Unit 1
NRC Project Manager, NRR – Three Mile Island Nuclear Station – Unit 1
NRC Project Manager, NMSS/DUWP/RDB – Three Mile Island – Unit 2
Director, Bureau of Radiation Protection - PA Department of Environmental Resources

Attachment 1

License Amendment Request
Three Mile Island Nuclear Station, Unit 1
Docket Nos. 50-289

EVALUATION OF PROPOSED CHANGES

Subject: Proposed Changes to Renewed Facility Operating License and Appendix A, Technical Specifications

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon), proposes an amendment to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI). The proposed license amendment request (LAR) would revise the RFOL and the associated TS to the Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. Exelon is also proposing changes to the Current Licensing Basis (CLB) mitigation strategies for Flood Mitigation and Aircraft Impact Protection in the Air Intake Tunnel.

By letter dated June 20, 2017 (Reference 1), Exelon provided formal notification to the U.S. Nuclear Regulatory Commission (NRC) of Exelon’s contingent determination to permanently cease operations at TMI on or about September 30, 2019. After docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel. As a result, TMI will be authorized to only possess special nuclear material. In support of this condition, the TMI RFOL and associated TS are being proposed for revision to reflect the planned permanent shutdown and defueled condition pursuant to 10 CFR 50.36(c)(6) "*Decommissioning*."

The proposed changes to the RFOL and TS are in accordance with 10 CFR 50.36(c)(1) through (c)(5). The proposed changes also include administrative changes to format (margins, font, tabs, etc.) of content, revised numbering of sections and pages; and the deletion of unused placeholders, where appropriate, to condense the number of pages in the TS without affecting the technical content. The TS Table of Contents is also revised accordingly.

The current TMI TS have been customized over the years to meet the specific needs of the unit. These TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including safe storage and management of irradiated fuel. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a permanently defueled plant.

In the development of the proposed PDTS changes, Exelon reviewed the PDTS requirements from other plants that have permanently shutdown, primarily Fort Calhoun (Reference 2), Vermont Yankee (Reference 3), Kewaunee (Reference 4), and Crystal River Nuclear Plant, Unit 3 (Reference 5). Exelon also evaluated the applicable guidance in NUREG-1430, *"Standard Technical Specifications Babcock and Wilcox Plants"* (Reference 6) and Draft NUREG-1625, *"Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants"* (Reference 7).

This LAR provides a discussion and description of the proposed RFOL, TS changes, and CLB changes with a technical evaluation of the proposed changes, and information supporting a finding of No Significant Hazards Consideration (NSHC).

Related Licensing Actions

By letter dated December 29, 2017 (Reference 8), the NRC approved the Certified Fuel Handler (CFH) Training and Retraining Program for TMI. By letter dated November 10, 2017 (Reference 9), Exelon submitted an LAR to the NRC proposing changes to the organization, staffing, and training requirements contained in TMI TS Section 6, Administrative Controls, which is incorporated into this LAR. The CFH program and the referenced LAR will become effective and will be implemented once the TMI reactor has been completely defueled and the certification of permanent removal of fuel from the reactor vessel has been docketed pursuant to 10 CFR 50.82(a)(1)(ii).

There are currently no other pending license amendment requests involving proposed changes to TS currently docketed for TMI. The above-mentioned licensing actions complement and support this proposed license amendment.

2.0 DETAILED DESCRIPTION AND BASIS FOR THE CHANGES

The proposed amendment would modify the TMI RFOL and transform the operating TMI TS into the TMI PDTS to comport with a permanently defueled condition, as well as clarifying current licensing bases to reflect the permanently defuel condition. To support the proposed changes, Exelon has evaluated the Design Basis Accidents (DBA) that will be applicable in a permanently shutdown and defueled condition. Exelon also evaluated the General Design Criteria (GDC) with respect to compliance in the permanently shutdown and defueled condition. The DBA and GDC evaluations provide the framework and basis for the proposed changes.

Design Basis Accident Analysis Applicable to Proposed Change

Chapter 14 of the TMI Updated Final Safety Analysis Report (UFSAR) contains the DBAs and transient scenarios applicable to TMI during power operations. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the reactor coolant system (RCS). Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core.

With the termination of reactor operations at TMI and the permanent removal of fuel from the reactor pressure vessel (RPV) as certified in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and pursuant to 10 CFR 50.82(a)(2), the majority of the DBA scenarios postulated in the UFSAR are no longer be possible. During decommissioning the irradiated fuel will be stored in the Spent fuel pool (SFP), and/or in the Independent Spent Fuel Storage Installation (ISFSI) once constructed, until it is shipped off site in accordance with the schedules to be provided in the Post Shutdown Decommissioning Activities Report (PSDAR) and the Spent Fuel Management Plan. The RCS, steam system, and turbine generator are no longer in operation and have no function related to the safe storage and management of the spent nuclear fuel.

Chapter 14 of the TMI UFSAR describes the safety analysis aspects of the plant that were evaluated to demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed the regulatory guidelines of 10 CFR 50.67, *"Accident source term,"* or 10 CFR Part 100, *"Reactor Site Criteria,"* as applicable. Two basic groups of events are pertinent to safety, which are abnormal operational transients and postulated DBAs; these two groups were investigated separately. The analyses of the abnormal operational transients evaluate the ability of the plant protection features to ensure that, during these transients, no fuel damage occurs and the RCS pressure limit is not exceeded. The safety design limits require that damage to the fuel be limited and that no nuclear system process barrier damage results from any abnormal operational occurrence. Thus, analysis of this group of events evaluates the features that protect the first two radioactive material barriers. Analysis of the events in the second group, postulated DBAs, evaluates situations that require functioning of the engineered safeguards in order to protect the fission product barriers, including containment, in order to minimize the offsite radiological consequences.

Safety analyses are integral of the plant's design and licensing basis. The safety analyses demonstrate the integrity of the fission product barriers, the capability to shutdown the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents and transients. Systems, Structures, and Components (SSCs) that perform design basis functions are credited in the safety analyses for the purpose of mitigating the transient or accident.

A list of the Chapter 14 DBAs and whether the accident applies to a permanently defueled condition is provided in Table 2.1.

TABLE 2.1 – TMI Design Basis Accidents

Postulated Accident or Transient	Defueled Applicability
14.1 Core and Coolant Boundary Protection Analysis	
14.1.1 Abnormalities	Not Applicable
14.1.2 Analysis of Effects and Consequences	
14.1.2.1 Uncompensated Operating Reactivity Changes	Not Applicable
14.1.2.2 Startup Accident	Not Applicable
14.1.2.3 Rod Withdrawal Accident at Rated Power Operation	Not Applicable
14.1.2.4 Moderator Dilution Accident	Not Applicable
14.1.2.5 Cold Water Accident	Not Applicable
14.1.2.6 Loss of Coolant Flow	Not Applicable
14.1.2.7 Stuck-Out, Stuck-In, Or Dropped Control Rod Accident	Not Applicable
14.1.2.8 Loss of Electric Power	Not Applicable
14.1.2.9 Steam Line Break	Not Applicable
14.1.2.10 Steam Generator Tube Failure	Not Applicable
14.1.2.11 Anticipated Transients Without Scram (ATWS)	Not Applicable
14.2 Standby Safeguards Analysis	
14.2.1 Situations Analyzed and Causes	Not Applicable
14.2.2 Accident Analyses	
14.2.2.1 Fuel Handling Accident	Applicable
14.2.2.2 Rod Ejection Accident	Not Applicable
14.2.2.3 Large Break Loss of Coolant Accident	Not Applicable

14.2.2.4 Small Break Loss of Coolant Accident	Not Applicable
14.2.2.5 Maximum Hypothetical Accident	Not Applicable
14.2.2.6 Waste Gas Tank Rupture	Applicable
14.2.2.7 Loss of Feedwater Accident	Not Applicable
14.2.2.8 Fuel Cask Drop Accident	Applicable
14.2.2.9 Feedwater Line Break Accident	Not Applicable

The accidents that remain applicable to TMI in the permanently shut down and defueled condition are the Fuel Handling Accident (FHA) within the SFP, the Waste Gas Tank Rupture (WGTR) and the Fuel Cask Drop Accident (FCDA). The Fuel Handling Accident within the reactor building is no longer an applicable concern. The potential accidents have been re-evaluated for the permanently defueled condition where all fuel has been removed from the reactor building.

Fuel Handling Accident Analysis for the Permanently Defueled Condition

A new FHA analysis in the SFP area for the permanently defueled condition has been completed, "Fuel Handling Accident Dose Consequence (Post Permanent Shutdown)" (Reference 10). The resultant dose consequences have been determined to remain within the limits of 10 CFR 50.67 and Regulatory Guide 1.183 (Reference 11).

The Post Permanent Shutdown FHA was evaluated using the methodology described in Regulatory Guide 1.183. This new analysis did not credit the function of any SSC or active mitigation measures. The analysis credits the decontamination of the 23 feet of water over the fuel assemblies in the SFP (i.e., 99.5% (or a Decontamination Factor (DF) of 200) of the iodine released from the fuel assembly is assumed to remain in the water).

The FHA is defined as the dropping of a single spent fuel assembly in the SFP during fuel handling activities, such that the entire outer row of fuel rods in the assembly, 56 of 208, suffers mechanical damage to the cladding. This accident is postulated to occur despite the administrative controls and physical limitations imposed on fuel-handling operations. The gap activity in the damaged rods is instantaneously released into the SFP. The release occurs under 23 feet of water, which acts as a filter.

Fuel release fractions from RG 1.183, Table 3 are doubled in this analysis in order to bound fuel assemblies potentially exceeding the RG 1.183, footnote 11 value of 6.3 kW/ft. peak rod average power for burnups exceeding 54 GWD/MTU. This is consistent with the licensing basis FHA analysis that was approved by the NRC in Reference 12. Additionally, the krypton (Kr)-85 and iodine (I)-131 inventories are increased by factors of 2 and 1.6, respectively, in order to account for additional fractional increases relative to other noble gas and iodine isotopes. At 60 days after reactor shutdown, damage to the fuel assembly of the highest activity results in 535 Ci of noble gas (adjusted Kr-85) and 0.86 Ci of iodine (adjusted I-131) being released from the SFP. These values were used in the Post Permanent Shutdown FHA analysis (Reference 10). Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed.

The activity released is assumed to be reach the environment outside the building within two hours. The Auxiliary and Fuel Handling Building Ventilation systems exhaust discharge to the atmosphere at the top of the reactor building; however, conservative atmospheric dispersion coefficients based on a ground level release are applied. The Post Permanent Shutdown FHA analysis does not take credit for: (1) the Fuel Handling Building emergency ventilation system, (2) Fuel Handling Building isolation during fuel movements, and (3) control room filtration and ventilation via the normal and emergency control room ventilation systems. Additionally, no control room isolation or recirculation/filtration is assumed in this analysis. It is assumed that the control room normal ventilation system fails after the onset of the accident resulting in control room isolation and the subsequent 'trapping' of unfiltered intake air in the control room. A sensitivity study was performed to determine the limiting dose consequences based on control room isolation time. After the control room is isolated a flow rate of 10 cfm into and out of the control room is assumed to account for normal ingress and egress.

Without crediting mitigation by any active SSC, the dose consequence of the Post Permanent Shutdown FHA at 60 days after reactor shutdown (i.e. unmitigated release of the radioisotope inventory from a FHA in the SFP) is as follows:

Location	Dose Limits	Dose Analysis Results
CR - Control Room	5.0 Rem	3.00 Rem TEDE
Exclusion Area Boundary	6.3 Rem	8.26 E-3 Rem TEDE
Low Population Zone	6.3 Rem	1.45 E-3 Rem TEDE

The Control Room, EAB and LPZ dose consequences from a Post Permanent Shutdown FHA will remain within regulatory limits 60 days after shutdown without crediting any active SSC. To maintain the assumptions of the Post Permanent Shutdown FHA analysis, Exelon proposes a new LCO 3.1.1 to ensure the minimum water level (23 feet above irradiated fuel assemblies) in the SFP is established prior to fuel handling and maintained during fuel handling evolutions.

<u>Waste Gas Tank Rupture</u>

Following permanent shutdown, the waste gas tanks will be required to retain and release waste gas generated from water management activities for a limited duration. The WGTR DBA (UFSAR 14.2.2.6) assumes that the Waste Gas Tank contains the gaseous activity evolved from degassing all of the RCS following operation with one percent defective fuel. The WGTR analysis assumes the waste gas is released to the Auxiliary Building and then to the environment as an instantaneous puff release. No credit is taken for any active safety system for the mitigation of the accident.

The Waste Gas Disposal System collects, stores, monitors, sample and releases radioactive gas, hydrogen and oxygen from the primary coolant. The source term contained in the waste gas tanks is based on the activity of the primary coolant. Once the reactor is permanently shutdown and defueled there is no mechanism to raise the primary coolant activity. Therefore, upon permanent shutdown and cooldown, the source term contained within the waste gas tanks represents the highest (worst case) source term and is expected to be significantly less than the assumed WGTR analysis. Subsequent additions to the waste gas tanks resulting from water management activities would be less than the final shutdown and cooldown waste gas tank source term.

The WGTR analysis as described in the UFSAR remains valid after permanent defueling; however, control room dose impact was not evaluated since it was bounded by other events. The Post Permanent Shutdown FHA analysis demonstrates that the unmitigated release of 535 Ci of Noble Gas and 0.86 Ci of iodine does not exceed 10 CFR 50.67 limits. The release paths for both analyses are consistent. If the activity in a waste gas tank is less than the activity predicted to be released in the Post Permanent Shutdown FHA analysis, then the resultant doses will remain within established limits. Prior to implementing the PDTS, the waste gas tank concentration will be measured and verified to be less than above values.

Based on the decreasing activity to the waste gas tanks after permanent defueling and verifying that the waste gas tank activity is less than that in the Post Permanent Shutdown FHA analysis, ensures that the result of a WGTR would not exceed control room dose limits.

Fuel Cask Drop Accident

The fuel cask drop accident (UFSAR 14.2.2.8) is defined as the dropping of a fuel cask through the maximum drop height during transfer operations of fuel cask onto a rail car. The FCDA analysis assumes the release of the noble gas and iodine is directly to the atmosphere and occurs instantaneously. No credit is taken for any active safety system for the mitigation of the accident.

The analysis of FCDA as described the UFSAR remains valid; however, it is unlikely that the fuel cask to which this applies will be used. Since TMI does not currently have an ISFSI, no casks are anticipated to be loaded until after the ISFSI is completed in 2021.

DBA Conclusion

The remaining DBAs that support permanently shutdown and defueled condition do not rely on any active safety system for mitigation. The new FHA analysis demonstrates that the unmitigated release of 535 Ci of Noble Gas and 0.86 Ci of iodine will not exceed 10 CFR 50.67 limits for the Control Room, or the doses at the EAB and LPZ. The following items are assumed:

- Proposed Technical Specification LCO 3.1.1 will ensure the minimum water level in the spent fuel pool is established prior to fuel handling and maintained.
- The activity within any waste gas tank will be confirmed to less the 535 Ci of Noble Gas and 0.86 Ci of iodine prior to implementation of the PDTS.

Detailed Review of General Design Criteria After Permanent Defueling

As discussed in UFSAR Section 1.4, Three Mile Island Nuclear Station, Unit 1 was designed and constructed taking into consideration the GDC for nuclear power plant construction permits as listed in the proposed Atomic Energy Commission (AEC) General Design Criteria, dated July 1967. Compliance with each of the proposed AEC criterion was reviewed as applied to the permanently shutdown and defueled condition and the limitations imposed by 10 CFR 50.82(a)(2) upon docketing the certification required by 10 CFR 50.82(a)(1)(ii). This review determined that compliance could be expressed in four categories as follows:

A. No Longer Applies – Compliance with the TMI GDC is no longer applicable since the intent and scope are based on conditions in which the unit can no longer be placed. Based on the permanently defueled condition the facility is no longer permitted to operate or emplace or retain fuel in the reactor vessel. The DBAs that evaluate conditions applicable to operation of the reactor no longer apply. Based on the DBAs bounding the safe handling and storage of fuel in the SFPs, no credit was taken for any active safety system for the mitigation of an analyzed event.

B. Unchanged – Compliance with the TMI GDC remains as stated in the UFSAR. The scope and intent of the GDC is not impacted by the transition from operating status to permanently shutdown and defueled status.

C. Minor Changes – Compliance with the TMI GDC is still required; however, the scope can be reduced based on the transition from operating status to permanently shutdown and defueled status.

D. Major Changes – Compliance with the TMI GDC is still required; however, the intent and scope are impacted by the transition from operating status to permanently shutdown and defueled status. These are discussed in further detail below, to reflect the proposed change to the TMI licensing bases.

A list of the current GDC and the impact to a permanently defueled condition based on the above dispositions are provided in Table 2.2.

Table 2.2 − TMI Compliance with AEC Proposed Design Criteria

AEC General Design Criteria		Defueled Applicability
Criterion 1	Quality Standards	D. Major Change
Criterion 2	Performance Standards	D. Major Change
Criterion 3	Fire Protection	D. Major Change
Criterion 4	Sharing of Systems	B. Unchanged
Criterion 5	Records Requirements	D. Major Change
Criterion 6	Reactor Core Design	A. No Longer Applies
Criterion 7	Suppression of Power Oscillations	A. No Longer Applies
Criterion 8	Overall Power Coefficient	A. No Longer Applies
Criterion 9	Reactor Coolant Pressure Boundary	A. No Longer Applies
Criterion 10	Containment	A. No Longer Applies
Criterion 11	Control Room	D. Major Change
Criterion 12	Instrumentation and Control Systems	A. No Longer Applies
Criterion 13	Fission Process Monitors and Controls	A. No Longer Applies
Criterion 14	Core Protection Systems	A. No Longer Applies
Criterion 15	Engineered Safety Features Protection Systems	A. No Longer Applies
Criterion 16	Monitoring Reactor Coolant Pressure Boundary	A. No Longer Applies
Criterion 17	Monitoring Radioactivity Release	C. Minor Change
Criterion 18	Monitoring Fuel and Waste Storage	B. Unchanged
Criterion 19	Protection Systems Reliability	A. No Longer Applies
Criterion 20	Protection Systems Redundancy and Independence	A. No Longer Applies
Criterion 21	Single Failure Definition	A. No Longer Applies
Criterion 22	Separation of Protection and Control Instrumentation Systems	A. No Longer Applies
Criterion 23	Protection Against Multiple Disability for Protection Systems	A. No Longer Applies
Criterion 24	Emergency Power for Protection Systems	A. No Longer Applies

AEC General Design Criteria		Defueled Applicability
Criterion 25	Demonstration of Functional Operability of Protection Systems	A. No Longer Applies
Criterion 26	Protection Systems Fail-Safe Design	A. No Longer Applies
Criterion 27	Redundancy of Reactivity Control	A. No Longer Applies
Criterion 28	Reactivity Hot Shutdown Capability	A. No Longer Applies
Criterion 29	Reactivity Shutdown Capability	A. No Longer Applies
Criterion 30	Reactivity Holddown Capability	A. No Longer Applies
Criterion 31	Reactivity Control Systems Malfunction	A. No Longer Applies
Criterion 32	Maximum Reactivity Worth of Control Rods	A. No Longer Applies
Criterion 33	Reactor Coolant Pressure Boundary Capability	A. No Longer Applies
Criterion 34	Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention	A. No Longer Applies
Criterion 35	Reactor Coolant Pressure Boundary Brittle Fracture Prevention	A. No Longer Applies
Criterion 36	Reactor Coolant Pressure Boundary Surveillance	A. No Longer Applies
Criterion 37	Engineered Safety Features Basis for Design	A. No Longer Applies
Criterion 38	Reliability and Testability of Engineered Safety Features	A. No Longer Applies
Criterion 39	Emergency Power for Engineered Safety Features	A. No Longer Applies
Criterion 40	Missile Protection	A. No Longer Applies
Criterion 41	Engineered Safety Features Performance Capability	A. No Longer Applies
Criterion 42	Engineered Safety Features Components Capability	A. No Longer Applies
Criterion 43	Accident Aggravation Prevention	A. No Longer Applies
Criterion 44	Emergency Core Cooling Systems Capability	A. No Longer Applies
Criterion 45	Inspection of Emergency Core Cooling Systems	A. No Longer Applies
Criterion 46	Testing of Emergency Core Cooling Systems Components	A. No Longer Applies
Criterion 47	Testing of Emergency Core Cooling Systems	A. No Longer Applies
Criterion 48	Testing of Operational Sequency of Emergency Core Cooling Systems	A. No Longer Applies
Criterion 49	Containment Design Basis	A. No Longer Applies
Criterion 50	NDT Requirement for Containment Material	A. No Longer Applies
Criterion 51	Reactor Coolant Pressure Boundary Outside Containment	A. No Longer Applies
Criterion 52	Containment Heat Removal Systems	A. No Longer Applies
Criterion 53	Containment Isolation Valves	A. No Longer Applies

AEC General Design Criteria		Defueled Applicability
Criterion 54	Containment Leakage Rate Testing	A. No Longer Applies
Criterion 55	Containment Periodic Leakage Rate Testing	A. No Longer Applies
Criterion 56	Provisions for Testing of Penetrations	A. No Longer Applies
Criterion 57	Provisions for Testing of Isolation Valves	A. No Longer Applies
Criterion 58	Inspection of Containment Pressure-Reducing Systems	A. No Longer Applies
Criterion 59	Testing of Containment Pressure-Reducing System Components	A. No Longer Applies
Criterion 60	Testing of Containment Spray Systems	A. No Longer Applies
Criterion 61	Testing of Operational Sequence of Containment Pressure-Reducing Systems	A. No Longer Applies
Criterion 62	Inspection of Air Cleanup Systems	A. No Longer Applies
Criterion 63	Testing of Air Cleanup Systems Components	A. No Longer Applies
Criterion 64	Testing of Air Cleanup Systems	A. No Longer Applies
Criterion 65	Testing of Operational Sequence of Air Cleanup Systems	A. No Longer Applies
Criterion 66	Prevention of Fuel Storage Criticality	B. Unchanged
Criterion 67	Fuel and Waste Storage Decay Heat	B. Unchanged
Criterion 68	Fuel and Waste Storage Shielding	B. Unchanged
Criterion 69	Protection Against Radioactivity Release from Spent Fuel and Waste Storage	B. Unchanged
Criterion 70	Control of Releases of Radioactivity to the Environment	B. Unchanged

Criterion 1 - Quality Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality Assurance programs, test procedures, and inspection acceptance levels used shall be identified. A showing of sufficiency and applicability of codes, standards, Quality Assurance programs, test procedures, and inspection acceptance levels used is required.

Discussion:

The TMI Quality Assurance Program will be contained in the Exelon Decommissioning Quality Assurance Program (DQAP). The DQAP is based on the existing Exelon Fleet Quality Assurance Topical Report (QATR) that is currently used at all operational Exelon plants. The DQAP reflects changes and simplifications based on site's decommissioning status. The DQAP was approved by the NRC on June 27, 2018 (Reference 13). The DQAP provides a top-level overview of the quality assurance program controls applied to quality related items and activities at Exelon plants during the decommissioning phase of the plant life. The DQAP is based on the applicable portions of 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix B to Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; 10 CFR Part 71, "Packaging and transportation of radioactive material," Subpart H, "Quality Assurance"; and 10 CFR Part 72, "Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related greater than Class C waste," Subpart G, "Quality Assurance."

Criterion 2 - Performance Standards

Those systems and components of Reactor Building facilities which are essential to the prevention of accidents, which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornados, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (1) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area, and (2) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Discussion:

The systems and components essential to public safety have been designed to performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces or effects which might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data, or upon the most severe conditions established based on synthetic analyses. In the permanently defueled condition, the Reactor Building is no longer a primary concern. The focus is shifted to the spent fuel contained in the SFP and the external event hazards applicable are described below:

a. Earthquakes: SSCs which maintain the integrity of the SFP are designed and maintained to Seismic Class I requirements to ensure the water inventory of the SFP is not lost following the maximum expected earthquake (UFSAR 2.8.2). This volume of water provides a reliable margin to restore active means of cooling and prevent fuel damage (Reference 14). No active function of an SSC must be maintained to protect the public following the maximum expected earthquake.

b. Tornadoes: The SFP is located within the Fuel Handling Building, which is designed and maintained to withstand the wind and missile hazards of the design basis tornado (UFSAR 5.2.1.6). This ensures the water inventory of the SFP is not lost. This volume of water provides a reliable margin to restore active means of cooling and prevent fuel damage (Reference 14). No active function of an SSC must be maintained to protect the public following a design basis tornado.

c. Floods: The SFP is located within the Fuel Handling Building, which is designed and maintained to withstand the hydraulic and debris forces of the Probable Maximum Flood

(PMF) (UFSAR 2.6.4). This ensures the water inventory of the SFP is not lost. The period of inundation for the PMF is less than 50 hours (UFSAR Figure 2.6-9). This volume of water provides a reliable margin to restore active means of cooling and prevent fuel damage (Reference 14). No active function of an SSC must be maintained to protect the public following a PMF.

d. Aircraft Impact: The SFP is located within the Fuel Handling Building, which is designed and maintained to withstand the postulated impact of an aircraft (UFSAR Appendix 5A). This ensures the water inventory of the SFP is not lost. This volume of water provides a reliable margin to restore active means of cooling and prevent fuel damage (Reference 14). No active function of an SSC must be maintained to protect the public following a design aircraft impact event.

e. Winds: Wind loads from events other than a tornado are less. The design for the tornado ensures the design is adequate for other wind related events.

f. Snow and Ice: The SFP is located within the Fuel Handling Building, which is designed and maintained to withstand the design snow and ice loads (UFSAR Appendix 5.2.1.2.5).

<u>Criterion 3 - Fire Protection</u>

The reactor facility shall be designed: (1) to minimize the probability of events such as fires and explosions, and (2) to minimize the potential effects of such events on safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, Control Room, and components of engineered safety features.

Discussion:

Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," describes the changes of the fire protection program related to operating unit as required by GDC 3 transitioning to a permanently shutdown condition.

> *“The primary objectives of the fire protection program for operating reactors are to minimize fire damage to structures, systems, and components (SSCs) important to safety; to ensure the capability to safely shut down the reactor; and to maintain it in a safe shutdown condition. For an initial period following shutdown, accidents that can challenge the 10 CFR Part 100 limits remain credible. The fire protection program should continue to provide protection against these events. The primary fire protection concern for permanently shutdown plants is protecting the integrity of the spent fuel and preventing or minimizing the release of radioactive materials resulting from fires involving contaminated plant SSCs or radioactive wastes. The radiation dose limits specified in 10 CFR Part 20, "Standards for Protection Against Radiation," apply to plant personnel and members of the public for fire incidents at permanently shutdown nuclear power plants. Licensees should make every effort to maintain exposures to radiation resulting from a fire as low as reasonably achievable.”*

The fire protection program for a decommissioned unit are governed by the requirements of 10 CFR 50.48(f).

Criterion 5 – Records Retention

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the licensee or under corporate control throughout the life of the reactor.

Discussion:

Records will continue to be retained at TMI as address by the Decommissioning Quality Assurance Program (10 CFR 50, Appendix B, Criterion XVII). Exelon intends to submit an exemption request for TMI to allow the reduction of certain regulatory required record retention requirements, consistent with similar exemption requests that have been approved by the NRC for other nuclear power reactor facilities beginning decommissioning. The exemption would include: (1) records associated with SSCs and activities that were applicable to the nuclear unit, which are no longer required by the 10 CFR Part 50 licensing basis (i.e., removed from the updated final safety analysis report and/or technical specifications by appropriate change mechanisms; and (2) records associated with the storage of spent nuclear fuel in the SFP once all fuel has been removed from the SFP and the TMI license no longer allows storage of fuel in the SFP.

Criterion 11 - Control Room

The facility shall be provided with a Control Room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the Control Room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposure of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the Control Room is lost due to fire or other cause.

Discussion:

Safe occupancy of the Control Room during abnormal conditions has been provided for in the original design. The Control Room is located in a Class I structure which was designed for the hypothetical aircraft incident. Adequate shielding has been provided to maintain tolerable radiation levels in the Control Room even in the event of a maximum hypothetical accident. The DBAs related to safe handling and storage of irradiated fuel in the SFP do not rely on an active safety system for mitigation. There are no immediate actions required to be conducted in the control room to respond to a loss of SFP cooling abnormal event.

The TS LAR, submitted November 10, 2017 (Reference 9), addressing proposed changes to Section 1 and Section 6 of the TMI TS, states that the command function would maintained at the location of the Unit's Shift Manager, and that the requirement to maintain a licensed operator in the Control Room would no longer be required. This submittal recognized that the Control Room will remain the physical center of the command function; however, associated activities (e.g. fuel handling or SFP cooling) do not necessarily rely on the control room and control activities may be performed either remotely from the control room or locally in the plant.

Inaccessibility of the Control Room is considered to be very improbable; nevertheless, loss of access or loss of control room functions does not pose a threat to public safety.

Detailed Discussion of Revisions to Current Licensing Basis

Revision to Flood Mitigation Strategy

The proposed flood protection strategy after permanent defueling will no longer credit or be dependent upon the Flood Barrier System (FBS) to ensure SFP cooling is maintained in the event of a PMF.

The design flood described in the TMI license basis is a Susquehanna River peak flow of 1,100,000 cubic feet per second (CFS). This event produces a peak water level of 301.6' elevation. The TMI site is elevated above this height and is surrounded by an earthen barrier (i.e., dike) which would prevent inundation of the site for river levels up to 304' elevation. Due to a change in the Susquehanna River PMF during the original licensing process, TMI committed to provide for a safe and orderly shutdown for the revised PMF (LCO 3.14.2). The PMF is an event with a Susquehanna River peak flow of 1,625,000 CFS, a warning time of at least 30 hours, a peak river water level of 313.3' elevation, and a period of inundation of 50 hours. The FBS (i.e., a system of seals, hatches and flood gates, and safety related structures described in UFSAR Section 2.6.5) ensures the function of safe shutdown systems could be maintained throughout a PMF event.

After the reactor is permanently shut down and defueled, the potential for an external flood event to cause a radiological release approaching regulatory limits is significantly reduced. The integrity of the Fuel Handling Building is not impacted by the PMF. In response to post Fukushima actions, the external flood hazard was reevaluated and mitigation capability was enhanced. The reevaluated flood hazard (Reference 21) was found to be less severe than the original license basis PMF. However, new equipment (diesel generators and pumps) was installed which provides an indefinite capability to maintain spent fuel cooling with a peak river water level at 320’ elevation. The SFP FLEX mitigation strategy will be maintained post permanent shutdown and defueling.

If a PMF occurred and the dike and FBS were not available, the normal means of SFP cooling would be lost, but the integrity of the SFP would not be adversely affected. The plant staff would have at least 13 hours before pool boiling occurred and more than 7 days to restore spent fuel cooling before fuel damage or a significant radiological release could occur. This time is more than adequate to reliably restore spent fuel cooling using the redundant components installed for post Fukushima flood protection or equipment obtained from offsite if necessary. (Reference 14)

Revision to Protection from Aircraft Impact

The aircraft impact hazard and design protection is described in the TMI license basis in UFSAR Sections 1.2.7, 5.1.3, and 9.9.6. There are no TS LCO's associated with protection for aircraft impact; however, an administrative control, TS 6.9.1 requires an annual report to provide the total number of aircraft movements to the NRC Region I Administrator. After the reactor is permanently shut down and all fuel is in the SFP, the potential for an aircraft impact event to cause a radiological release approaching 10 CFR 100 limits is very low (below the Standard Review Plan (SRP) guideline for external hazards) and the control room is not needed to conduct mitigative action to respond to such an event.

The proposed license amendment eliminates the licensing basis requirements in the current USFAR for the automatic suppression of an explosion or fire in the Air Intake Tunnel (AIT) and the general requirements for automatic ventilation shutdown or isolation of sump flow paths if combustible vapors are present. Specifically, the UFSAR description of aircraft impact design protection would be revised to not include automatic AIT isolation or general combustible vapor design features in the event of an aircraft impact. Smoke detection and manual fire water deluge will be maintained for the AIT according to the fire program requirements.

An aircraft impact on the Fuel Handling Building would not have any adverse effect on SFP integrity or the ability to timely restore spent fuel level/cooling. If an aircraft did strike the Air Intake Pagoda and the automatic detection and suppression were not available, then a fire or explosion in the AIT could occur. The fire would be detected by smoke detectors which alarm in the control room. The fire water deluge system for the AIT would be manually actuated to suppress the fire. These design features may not be sufficient to prevent damage to electrical cables in the tunnel or maintain control room habitability. Those failures could interrupt the normal means of SFP cooling, but the integrity of the SFP would not be affected and SFP cooling could be manually restored from outside the control room. The plant staff would have at least 13 hours before pool boiling occurred and more than 7 days to restore spent fuel cooling before fuel damage or a significant radiological release could occur (Reference 14). This amount of time is more than adequate to manually restore spent fuel cooling using indications and controls from outside the control room. Additionally, this would allow sufficient time to retrieve and set-up either of the two redundant sets of post-Fukushima spent fuel cooling components (which are tested periodically) stored in the FLEX Storage Facility (an aircraft hardened structure more than 300 feet from the Air Intake Pagoda), or should provide sufficient time to obtain equipment from offsite, if necessary.

Detailed Discussion of Proposed RFOL and TS Changes

The following tables identify each RFOL and TS section that is being changed, the proposed change, and the basis for each change. Changes to the RFOL are addressed first, followed by the TS. Proposed revisions are shown in ***Bold-Italics*** and deletions are shown using *italicized ~~strikethrough~~*.

Attachment 2 provides the marked-up version of the TMI RFOL and TS. The TS that are deleted in their entirety are identified as such below, but the associated deleted pages are not included in Attachment 2. Proposed changes to the TS Bases addressing the proposed changes to the relevant TS are provided for information in Attachment 2. Upon approval of this amendment, changes to the Bases will be incorporated in accordance with TS 6.18, "Technical Specifications (TS) Bases Control Program," which is retained in its entirety without change.

Additionally, the proposed changes to the TS are considered a major rewrite. Revised formatting (margins, font, tabs, etc.) of content is used to create a continuous electronic file, revised numbering of sections and pages; and the deletion of unused placeholders, where appropriate, is used to condense and reduce the number of pages in the TS without affecting the technical content. Since the changes to the TS are considered a major rewrite, revision bars are not used. The TS Table of Contents is revised to reflect the remaining applicable sections and new page numbering. These changes are considered administrative and are shown in the marked-up pages (Attachment 2).

10 CFR 50.36, *"Technical specifications,"* promulgates the regulatory requirements related to the content of Technical Specifications. As detailed in subsequent sections of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS (see Section 3.1). In a permanently defueled condition, the scope of

equipment and parameters that must be included in the TMI PDTS is limited to those needed to address the remaining applicable DBAs so that the consequences of the accident are maintained within acceptable limits.

<table>
<tr><td colspan="2">RENEWED FACILITY <s>OPERATING</s> LICENSE
Formatting changes to this section will remove the header and footer titles and revise enclosure title to remove the use of "operating" to reflect the change from an operating license to being prohibited from operating the reactor pursuant to 10 CFR 50.82(a)(2). Pursuant to 10 CFR 50.51(b)(1), Exelon will continue to “maintain[ing] the facility” including “the storage, control and maintenance of the spent fuel, in a safe condition.”</td></tr>
<tr><td colspan="2">License Finding 1.b.</td></tr>
<tr><td><u>Current License Finding 1.b.</u>
Construction of the Three Mile Island Nuclear Station, Unit 1 (TMI or the facility) has been substantially completed in conformity with Construction Permit No: CPPR-40, the application, as amended, the provisions of the Act and the rules and regulations of the Commission;</td><td><u>Proposed License Finding 1.b.</u>
DELETED</td></tr>
<tr><td colspan="2">Basis</td></tr>
<tr><td colspan="2">This license finding is proposed for deletion in its entirety. Decommissioning of TMI is not dependent on the regulations that govern construction of the facility.</td></tr>
</table>

<table>
<tr><td colspan="2">License Finding 1.c.</td></tr>
<tr><td><u>Current License Finding 1.c.</u>
The facility will operate in conformity with the application, as amended, the provisions of the Act and the rules and regulations of the Commission;</td><td><u>Proposed License Finding 1.c.</u>
The facility will be maintained <s>operate</s> in conformity with the application, as amended, the provisions of the Act and the rules and regulations of the Commission;</td></tr>
<tr><td colspan="2">Basis</td></tr>
<tr><td colspan="2">This license finding is proposed for revision to reflect that the license no longer authorizes operation of the reactor. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with the decommissioning plant.</td></tr>
</table>

License Finding 1.d.	
Current License Finding 1.d. There is a reasonable assurance: (1) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted in compliance with the rules and regulations of the Commission;	Proposed License Finding 1.d. There is a reasonable assurance: (1) that the activities authorized by this renewed ~~operating~~ license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted in compliance with the rules and regulations of the Commission;
Basis	
This license finding is proposed for revision to reflect that the license no longer authorizes operation of the reactor. Once TMI has permanently ceased operation and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2). The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with the decommissioning plant.	

License Finding 1.g.	
Current License Finding 1.g. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;	Proposed License Finding 1.g. The issuance of this renewed ~~operating~~ license will not be inimical to the common defense and security or to the health and safety of the public;
Basis	
This license finding is proposed for revision to reflect that the license no longer authorizes operation of the reactor. Once TMI has permanently ceased operation and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2). The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the change is consistent with the requirements associated with the decommissioning plant.	

License Finding 1.h.	
Current License Finding 1.h. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Renewed Facility Operating License No. DPR-50 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;	Proposed License Finding 1.h. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Renewed Facility ~~Operating~~ License No. DPR-50 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;
Basis	
This license finding is proposed for revision to reflect that the license no longer authorizes operation of the reactor. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with the decommissioning plant.	

License Condition 2.	
Current License Condition 2. 2. Renewed Facility Operating License No. DPR-50 is hereby issued to Exelon Generation Company to read as follows:	Current License Condition 2. 2. Renewed Facility ~~Operating~~ License No. DPR-50 is hereby issued to Exelon Generation Company to read as follows:
Basis	
This license condition is proposed for revision to reflect that the license no longer authorizes operation of the reactor. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with the decommissioning plant.	

License Condition 2.a	
Current License Condition 2.a This renewed license applies to the Three Mile Island Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility), owned and operated by Exelon Generation Company. The facility is located in Dauphin County, Pennsylvania, and is described in the "Updated Final Safety Analysis Report (UFSAR)" as supplemented and amended and the Environmental Report as supplemented and amended.	Current License Condition 2.a This renewed license applies to the Three Mile Island Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility), owned ~~*and operated*~~ by Exelon Generation Company. The facility is located in Dauphin County, Pennsylvania, and is described in the "Updated Final Safety Analysis Report (UFSAR)" as supplemented and amended and the Environmental Report as supplemented and amended.
Basis	
This license condition is proposed for revision to reflect that the license no longer authorizes operation of the reactor. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with the decommissioning plant.	

License Condition 2.b.(1)	
Current License Condition 2.b.(1) Exelon Generation Company, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility in accordance with the procedures and limitations set forth in this renewed license;	Proposed License Condition 2.b.(1) Exelon Generation Company, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess~~,~~ ***and*** use~~*, and operate*~~ the facility ***as required for fuel storage*** in accordance with the procedures and limitations set forth in this renewed license;
Basis	
This license condition is proposed for revision to reflect the change from an operating license to being prohibited from operating the reactor pursuant to 10 CFR 50.82(a)(2). As such, the facility would remain authorized to possess the existing spent fuel and use the systems required to support safe fuel storage (e.g., the SFP) during the decommissioning period, in accordance with the specified limitations for storage.	

License Condition 2.b.(2)

Current License Condition 2.b.(2)	Proposed License Condition 2.b.(2)
Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as reactor fuel, sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required for reactor operation;	Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to ~~receive,~~ possess ~~and use~~ at any time any byproduct, source and special nuclear material ***used previously*** as reactor fuel, sealed neutron sources ***used previously*** for reactor startup, ***as fission detectors, and*** sealed sources for reactor instrumentation ~~and radiation monitoring equipment calibration, and as fission detectors~~ ***and to possess and use at any time any byproduct, source and special nuclear material as sealed sources for radiation monitoring equipment calibration*** in amounts as required ~~for reactor operation~~;

Basis

The proposed change to this license condition removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel. It eliminates the reference to use of the SNM for reactor operations and limits the possession of SNM to SNM that was "used previously" as reactor fuel. Pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for TMI will no longer authorizes operation of the reactor. As such, TMI has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM that was "used previously" as reactor fuel is necessary as TMI currently possesses the reactor fuel that was used for the past operations of the reactor.

The proposed change deletes the language regarding receipt of sealed neutron sources for reactor startup and reactor instrumentation. This license condition is revised to reflect authorization only for continued possession of those sources used for reactor startups, produced as a byproduct, and those required for calibration. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the use of startup sources is no longer needed. Therefore, the changes are consistent with the requirements associated with the decommissioning plant. The use of sources for Radiation Monitoring will continue to be required.

License Condition 2.b.(4)

Current License Condition 2.b.(4)	Proposed License Condition 2.b.(4)
Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess at the TMI Unit 1 or Unit 2 site, but not separate, such byproduct and special nuclear materials as may be produced by the operation of either unit.	Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess at the TMI Unit 1 or Unit 2 site, but not separate, such byproduct and special nuclear materials ~~as may be~~ ***that were*** produced by the operation of either unit.

Basis

This license condition is proposed for revision to allow possession of byproduct and SNM that were produced during operation of the reactor, but not allow the separation of material that was produced by

operations of the reactor. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), this license condition is consistent with the requirements associated with the decommissioning plant.

License Condition 2.c.(1)	
Current License Condition 2.c.(1) Maximum Power Level Exelon Generation Company is authorized to operate the facility at steady-state reactor core power levels not in excess of 2568 megawatts thermal.	Proposed License Condition 2.c.(1) ***DELETED***
Basis	
This license condition is proposed for deletion in its entirety. The requirements associated with the plant's maximum power level is proposed for deletion since TMI has permanently ceased power operations. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the use of a power limit is no longer needed. Therefore, the changes are consistent with the requirements associated with the decommissioning plant.	

License Condition 2.c.(2)	
Current License Condition 2.c.(2) Technical Specifications The Technical Specifications contained in Appendices A, as revised through Amendment No. 293, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.	Proposed License Condition 2.c.(2) Technical Specifications The Technical Specifications contained in ~~Appendices~~***Appendix*** A, as revised through Amendment No. ~~293~~***[###]***, are hereby incorporated in the license. Exelon Generation Company shall ~~operate~~ ***maintain*** the facility in accordance with the ***Permanently Defueled*** Technical Specifications ***(PDTS)***.
Basis	
This license condition is proposed for revision to account for the permanently defueled condition of the facility and to incorporate the Permanently Defueled Technical Specifications (PDTS). The paragraph is revised to reflect the nomenclature change to more accurately describe the document. Also changed is the designation from operating to maintaining the facility, which describes the defueled condition in which the TMI license no longer allows the use of the facility for power operation as provided in 10 CFR 50.82(a)(2).	

License Condition 2.c.(4)	
<u>Current License Condition 2.c.(4)</u> <u>Fire Protection</u> Exelon Generation Company shall implement and maintain in effect all provisions of the Fire Protection Program as described in the Updated FSAR for TMI. Changes may be made to the Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. 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,	<u>Proposed License Condition 2.c.(4)</u> ***DELETED***

Basis
This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once TMI has permanently ceased operation and certified that fuel has been permanently removed from the reactor, the fire protection program will be revised to take into account the facility conditions and activities during decommissioning. TMI will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment. This license condition, which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48 with the ability to achieve and maintain safe shut down of the reactor in the event of a fire, will no longer be applicable at TMI. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during facility decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. The regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shut down and defueled facility is not needed.

<table>
<tr><th colspan="2">License Condition 2.c.(5)</th></tr>
<tr><td><u>Current License Condition 2.c.(5)</u>

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;

e. Procedures defining corrective actions of off control point chemistry conditions; and

f. A procedure identifying (1) the authority responsible for the interpretation of the data, and (2) the sequence and timing of administrative events required to initiate corrective action.</td><td><u>Proposed License Condition 2.c.(5)</u>

DELETED</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is proposed for deletion in its entirety. The secondary water chemistry monitoring program contains procedures, sampling points, and sampling frequencies associated with critical parameters of secondary water chemistry to inhibit Steam Generator (SG) tube degradation. With the plant in a permanently defueled state the postulated Steam Generator Tube Failure accident analyzed in UFSAR Chapter 14 is no longer credible. Therefore, the secondary water chemistry program which is designed to prevent steam generator tube degradation that may lead to the Steam Generator Tube Failure accident is no longer needed. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the program associated with this license condition is no longer applicable; therefore, the proposed deletion of this section its entirety is acceptable.</td></tr>
</table>

<table>
<tr><th colspan="2">License Condition 2.c.(18)</th></tr>
<tr><td><u>Current License Condition 2.c.(18)</u>

Upon implementation of Amendment No. 264 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Specification 4.12.1.5, in accordance with TS 6.20.c.(i), the assessment of CRE habitability as required by Specification 6.20.c.(ii), and the measurement of CRE pressure as required by Specification 6.20.d, shall be considered met. Following implementation:

(a) The first performance of Specification 4.12.1.5, in accordance with Specification 6.20.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of Specification 1.25, as measured from August 21, 2000, the date of the most recent successful tracer gas test, as stated in the December 9, 2003, letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability, Specification 6.20.c.(ii), shall be within 3 years, plus the 9-month allowance of Specification 1.25, as measured from August 21, 2000, the date of the most recent successful tracer gas test, as stated in the December 9, 2003, letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 6.20.d, shall be within 24 months, plus the 180 days allowed by Specification 1.25, as measured from December 9, 2006, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.</td><td><u>Proposed License Condition 2.c.(18)</u>

DELETED</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">This license condition is proposed for elimination in its entirety. The proposed change removes the requirements of TSTF-448 that involve assessing the CRE Habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0. These assessments were completed in</td></tr>
</table>

accordance with the schedule specified in the license condition. Since the requirements of this one-time license condition have been completed, this license condition may be eliminated.

This submittal also proposes to eliminate TS 3.15.1 for control room ventilation system and TS 6.20 for the CRE Habitability Program. Since TS 3.15.1 and TS 6.20 are no longer necessary, this license condition is no longer needed; therefore, the proposed deletion of this license condition is acceptable.

License Condition 2.c.(19)	
Current License Condition 2.c.(19) At the time of the closing of the transfer of TM1-1, and the respective license from AmerGen Energy Company, LLC (AmerGen) to Exelon Generation Company, AmerGen shall transfer to Exelon Generation Company ownership and control of AmerGen TMI NQF, LLC, and AmerGen Consolidation, LLC shall be merged into Exelon Generation Consolidation, LLC. Also at the time of the closing, decommissioning funding assurance provided by Exelon Generation Company, using an additional method allowed under 10 CFR 50.75 if necessary, must be equal to or greater than the minimum amount calculated on that date pursuant to, and required by 10 CFR 50.75 for TMI. Furthermore, funds dedicated for TMI prior to closing shall remain dedicated to TMI following the closing. The name of AmerGen TMI NQF, LLC shall be changed to Exelon Generation TMI NQF, LLC at the time of the closing.	Proposed License Condition 2.c.(19) ***DELETED***
Basis	
This license condition is proposed for deletion in its entirety. This license condition is a one-time condition that eliminates references to AmerGen Energy Company, LLC, and replaces them with references to Exelon Generation Company, LLC, to reflect the results of the license transfer. AmerGen transferred to Exelon Generation Company ownership and control of AmerGen Three Mile Island NQF, LLC, and AmerGen Consolidation, LLC merged into Exelon Generation Consolidation, LLC. On December 23, 2008, the NRC approved the transfer of license and ownership of TMI to Exelon (Reference 15). The name of AmerGen Three Mile Island NQF, LLC was changed to Exelon Generation Three Mile Island NQF, LLC at the time of the closing. In a letter dated March 31, 2009, Exelon reported to the NRC that the decommissioning trust agreements for TMI had been modified to reflect the change in license from AmerGen Energy Co., LLC to Exelon (Reference 16). The requirements of this one-time license condition have been completed; therefore, this license condition may be eliminated.	

License Condition 2.c.(20)	
Current License Condition 2.c.(20) The information in the UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be incorporated into the UFSAR no later than the next scheduled update required by 10 CFR 50.71(e) following the issuance of this renewed operating license. Until this update is complete, Exelon Generation Company may not make changes to the information in the supplement. Following incorporation into the UFSAR, the need for prior Commission approval of any changes will be governed by 10 CFR 50.59.	Proposed License Condition 2.c.(20) ***DELETED***
Basis	
This license condition was issued concurrent with the Renewed Facility Operating License on October 22, 2009. This license condition is described in Section 1.7 "Summary of Proposed License Conditions," of NUREG-1928, *"Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1"* issued October 2009 (Reference 17). This license condition is a one-time requirement to update the UFSAR to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update as required by 10 CFR 50.71(e) and allows changes to be made to that supplement under the provisions of 50.59 when the UFSAR update is completed. TMI UFSAR, Revision 20, updated the UFSAR to include the supplement (Appendix A) for the License Renewal Application (LRA) (ECR 10-00654) and the updated UFSAR has been submitted to the NRC (Reference 18). Exelon notified the NRC of the completion of this license condition in a letter dated April 11, 2014 (Reference 19). This license condition has been completed in its entirety and therefore is proposed for deletion.	

License Condition 2.c.(21)	
Current License Condition 2.c.(21) The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to and/or during the period of extended operation. The licensee shall complete these activities in accordance with Appendix A of NUREG-1928, “Safety Evaluation Report Related to the License Renewal of Three Mile Island, Unit 1,” dated, October 2009. The licensee shall notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection.	Proposed License Condition 2.c.(21) ***DELETED***
Basis	
This license condition was issued concurrent with the Renewed Facility Operating License on October 22, 2009. This license condition is described in Appendix A "Long term Commitments for License Renewal of TMI," of NUREG-1928, *"Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1"* issued October 2009 (Reference 17). This license condition is a one-time requirement for the licensee to notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection. Exelon notified the NRC of the completion of this license condition in a letter dated April 11, 2014 (Reference 19). This license condition has been completed in its entirety and therefore is proposed for deletion.	

Proposed License Condition 2.c.(22)	
Current License Condition 2.c.(22) [None]	Proposed License Condition 2.c.(22) ***Handling of irradiated fuel in the Spent Fuel Pool will not be permitted following implementation of the PDTS until a minimum of 60 days following the permanent shutdown.***
Basis	
Exelon is requesting this proposed License Condition such that initial system abandonment activities may be started expeditiously after the permanent removal of fuel from the reactor vessel. By applying this License Condition, Exelon will be able to remove the TS requirements associated with those systems that perform mitigative actions assumed in the CLB FHA by precluding the possibility of a FHA until after the assumed 60-day decay period assumed in the Post Permanent Shutdown FHA has elapsed. Once the reactor has been permanently defueled with all spent fuel placed in the SFP and the certifications submitted and docketed in accordance with 10 CFR 50.82, power operation or emplacement of fuel in the reactor will not be allowed. Therefore, all DBAs associated with power operations or fuel handling in the	

Reactor Building will no longer be applicable, which provides the basis for removal of the Safety Limits and most of the Limiting Conditions for Operation.

The deletion of the Air Filtration System LCOs in LCO 3.15 are based on the new Post Permanent Shutdown FHA analysis, which is described in the "Fuel Handling Accident Analysis for the Permanently Defueled Condition" section of this Attachment. This analysis removes credit for any of the requirements in LCO 3.15 during fuel handling activities. However, this analysis assumes the irradiated fuel has decayed for 60 days after permanent shutdown.

Once the core is permanently offloaded into the SFP, TMI does not plan to handle or move irradiated fuel until it is relocated to the ISFSI. Currently, TMI does not have an ISFSI, but is pursuing the design and installation of one which is projected to be completed in 2021. Movement of irradiated spent fuel in the SFP is not be expected to be required until after the completion on the ISFSI in 2021, which is beyond the assumed 60-day decay period.

The only conditions that would require movement of irradiated fuel prior to movement of fuel to the ISFSI would be if an irradiated fuel assembly would be found to be erroneously loaded in a location not permitted by TS 5.4.2.g and h. (These TS requirements are being preserved as TS LCO 3.1.3 in the proposed PDTS). As part of the normal fuel handling requirements, Exelon validates compliance with TS 5.4.2.g and TS 5.4.2.h. Validation of compliance with TS 5.4.2.g and TS 5.2.2.h after the fuel has been permanently located in the SFP ensures that no fuel movements would be required during the period between implementation of PDTS and when the "Fuel Handling Accident Dose Consequence (Post Permanent Shutdown)" becomes valid 60-days after permanent shutdown.

In order to implement the PDTS prior to the 60-day decay time assumed in the Post Permanent Shutdown FHA analysis, Exelon proposes to prohibit movement of spent fuel after the submittal of the certification of permanent removal of fuel from the reactor vessel until 60 days after permanent shutdown through the imposition of the proposed License Condition. This will effectively prevent a FHA from occurring until after the 60-day decay period has elapsed.

License Condition d.	
<u>Current License Condition d.</u> This license is effective as of the date of issuance and shall expire at midnight on April 19, 2034.	<u>Proposed License Condition d.</u> This license is effective as of the date of issuance and ~~shall expire at midnight on April 9, 2029~~***is effective until the Commission notifies the licensee in writing that the license is terminated.***
Basis	
The proposed change modifies this license condition to reflect the permanently defueled condition of the facility. Once TMI has permanently ceased operation and certified that fuel has been permanently removed from the reactor, reference to operation of the facility would be inconsistent with the provisions of 10 CFR 50.82(a)(2). This license condition is being revised to conform with 10 CFR 50.51, *"Continuation of license,"* in that, the license authorizes ownership and possession by Exelon until the Commission notifies the licensee in writing that the license is terminated.	

TS Section 1 – Definitions

TS Section 1 "Definitions," contains defined terms that are applicable to an operating plant throughout the TS and TS Bases. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The revision to the definitions identified below align with the permanently shutdown and defueled reactor conditions. Many of the definitions have not been proposed for inclusion in the PDTS since they are relevant to an operating reactor and are no longer used in the TS. The standard convention of indicating the defined term in ALL CAPITAL LETTERS throughout the TS has been adopted in the PDTS.

Definitions to be Maintained

1.~~28~~***2*** <u>CERTIFIED FUEL HANDLER</u>

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.~~29~~***3*** <u>NON-CERTIFIED OPERATOR</u>

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

1.~~3~~***4*** <u>OPERABLE</u>

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

1.~~2.12~~***5*** <u>STATION, UNIT, PLANT, AND FACILITY</u>

Station, unit, plant, and facility as used in these technical specifications all refer to TMI Unit 1.

Basis

The definitions for Certified Fuel Handler (CFH) and Non-Certified Operator (NCO) were submitted to the NRC in an LAR dated November 10, 2017 (Reference 9). The definitions are proposed to be renumbered from 1.28 (CFH) to 1.2 and 1.29 (NCO) to 1.3 to place them in alphabetic order with the remaining TS definitions. This action is editorial in nature.

The definitions for OPERBILITY and STATION, UNIT, PLANT, AND FACILITY are proposed to be renumbered from 1.3 to 1.4 and 1.2.12 to 1.5, respectively, to place them in alphabetic order with the remaining TS definitions. This action is editorial in nature.

Definitions Proposed for Addition

1.1 ***<u>ACTIONS</u>***

ACTIONS shall be that part of a Specification that prescribes required actions to be taken under designated Conditions within specified completion times.

Basis for Addition

The definition for "Actions" is being added in order to clarify a term used in remaining TS sections. The definition is based on the definition in NUREG-1430, *"Standard Technical Specifications Babcock and Wilcox Plants"* (Reference 6).

The definition is proposed to be numbered 1.1 to place it in alphabetic order with the remaining TS definitions. This action is editorial in nature.

Definitions Proposed for Relocation
1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM) The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.3 and 6.9.4. 1.16 PROCESS CONTROL PROGRAM (PCP) The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.
Basis for Deletion
These definitions are not proposed for inclusion in the PDTS. These definitions are proposed to be relocated to the ODCM.

Reactor Operating Condition and Power Distribution Definitions Proposed for Deletion
1.1 RATED POWER Rated power is a steady state reactor core output of 2568 MWt. 1.2 REACTOR OPERATING CONDITIONS 1.2.1 COLD SHUTDOWN The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and T_{ave} is no more than 200°F. Pressure is defined by Specification 3.1.2. 1.2.2 HOT SHUTDOWN The reactor is in the hot shutdown condition when it is subcritical by at least one percent delta k/k and T_{ave} is at or greater than 525°F. 1.2.3 REACTOR CRITICAL The reactor is critical when the neutron chain reaction is self-sustaining and Keff = 1.0. 1.2.4 HOT STANDBY The reactor is in the hot standby condition when all of the following conditions exist: a. T_{ave} is greater than 525°F b. The reactor is critical

c. Indicated neutron power on the power range channels is less than two percent of rated power

1.2.5 POWER OPERATION

The reactor is in a power operating condition when the indicated neutron power is above two percent of rated power as indicated on the power range channels.

1.2.6 REFUELING SHUTDOWN

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 REFUELING OPERATION

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 REFUELING INTERVAL

Time between normal refuelings of the reactor. This is defined as once per 24 months.

1.2.9 STARTUP

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.2.10 T_{ave}

T_{ave} is defined as the arithmetic average of the coolant temperatures in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made.

1.2.11 HEATUP - COOLDOWN MODE

The heatup-cooldown mode is the range of reactor coolant temperature greater than 200°F and less than 525°F.

1.6 <u>POWER DISTRIBUTION</u>

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100\left[\frac{\textit{Power in Any Core Quadrant}}{\textit{Average Power of All Quadrants}} - 1\right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 AXIAL POWER IMBALANCE

Axial power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the Reactor Protection System (RPS) using input from the power range channels.

Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

Basis for Deletion

These definitions are not proposed for inclusion in the PDTS since the terms are not used in any PDTS specification and do not apply to a facility in the permanently defueled condition. These terms are meaningful only to a reactor authorized to contain fuel, operate at power, and refuel. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2).

Instrumentation Definitions Proposed for Deletion

1.4 PROTECTION INSTRUMENTATION LOGIC

1.4.1 INSTRUMENT CHANNEL

An instrument channel is the combination of sensor, wires, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control, and/or protection. An instrument channel may be either analog or digital.

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is described in Section 7.1 of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as described in Section 7.1 of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as described in Section 7.1 of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one out of two times two logic. Each element of the one out of two times two logic is controlled by a separate set of two out of four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 INSTRUMENTATION SURVEILLANCE

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

1.5.2 CHANNEL TEST

A CHANNEL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practical to verify OPERABILITY, including alarm and/or trip functions.

1.5.3 CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

1.5.4 CHANNEL CALIBRATION

An instrument CHANNEL CALIBRATION is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

1.5.5 HEAT BALANCE CHECK

A HEAT BALANCE CHECK is a comparison of the indicated neutron power and core thermal power.

1.5.6 HEAT BALANCE CALIBRATION

A HEAT BALANCE CALIBRATION is an adjustment of the power range channel amplifiers output based on the core thermal power determination.

Basis for Deletion

These definitions are not proposed for inclusion in the PDTS since the terms are not used in any PDTS specification and do not apply to a facility in the permanently defueled condition. These terms are meaningful only to a reactor authorized to contain fuel and operate at power. There is no instrumentation credited in the analysis of the accidents that remain credible (i.e., the FHA in the SFP). Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The Engineered Safety Feature (ESF) equipment and RPS Instrumentation have no function in the permanently defueled condition.

Containment and Miscellaneous Definitions Proposed for Deletion

1.7 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel and emergency air locks are closed and sealed.

b. All passive Containment Isolation Valves (CIVs) and isolation devices, including manual valves and blind flanges, are closed as required by the “Containment Integrity Check List” attached to the operating procedure, "Containment Integrity and Access Limits." Normally closed passive CIVs may be unisolated intermittently under administrative control.

c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently and manual control of power-operated valves may be substituted for automatic control under administrative control.

d. The containment leakage determined at the last testing interval satisfies Specification 4.4.1

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

1.12 DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same thyroid dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

1.26 DOSE EQUIVALENT Xe-133

Dose Equivalent Xe-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT Xe-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12.

1.19 PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is required to purify the confinement.

1.21 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

1.22 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the AmerGen Energy Company, LLC, AmerGen Energy Company, LLC contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make.

1.23 SUBSTANTIVE CHANGES

SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Examples of non-substantive changes are: (1) correcting spelling; (2) adding (but not deleting) sign off spaces; (3) blocking in notes, cautions, etc.; (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications; and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.

Basis for Deletion

These definitions are not proposed for inclusion in the PDTS since the terms are not used in any PDTS specification.

Definition 1.7 - In the permanently defueled condition there will be no DBAs for which primary containment integrity will be required to mitigate the consequences.

Definition 1.8 - During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. The regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license.

Definitions 1.12 and 1.26 do not apply to a facility in the permanently defueled condition. These terms are currently used in TS LCO 3.1.4 and LCO 3.13 to express the specific activity limit from a mixture of iodine isotopes and contained in primary or secondary coolant, respectively. TS LCO 3.1.4 and LCO 3.13 are not proposed for inclusion in the PDTS. The specific activity limit is used as the basis in accident analysis involving primary and secondary coolant releases. Since accident conditions associated with the RCS will no longer apply to the permanently shut down and defueled facility, the definition is no longer meaningful.

Definition 1.19 - There are no purge activities credited in the analyses of the accidents that remain credible.

Definition 1.21 - The term is defined and codified in the applicable regulations (e.g. 10 CFR 50.72 and 10 CFR 50.73); therefore, the definition need not be repeated in the PDTS. Administrative Controls TS 6.6, "Reportable Events Action," was proposed to be deleted in the LAR dated November 10, 2017 (Reference 9).

Definition 1.22 - The term is defined in 10 CFR 20.1003; therefore, the definition need not be repeated in the PDTS.

Definition 1.23 – This term is not used in any PDTS specification.

Definition Placeholders Proposed for Deletion
1.9 DELETED 1.10 DELETED 1.11 DELETED 1.14 DELETED
Basis for Deletion
These placeholders are proposed to be removed due to the elimination of other definitions. This action is editorial in nature.

Other Miscellaneous Definitions Proposed for Deletion

1.13 <u>SOURCE CHECK</u>

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.17 <u>GASEOUS RADWASTE TREATMENT</u>

The GASEOUS RADWASTE TREATMENT SYSTEM is the system designed and installed to reduce radioactive gaseous effluent by collecting primary coolant system off gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.18 <u>VENTILATION EXHAUST TREATMENT SYSTEM</u>

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluent by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodine or particulates from the gaseous exhaust system prior to the release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEMS.

1.20 VENTING

VENTING is the controlled process of discharging air as gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is not provided. Vent used in system name does not imply a VENTING process.

1.24 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is a TMI specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these operating limits is addressed in individual specifications

1.27 INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

Basis for Deletion

These definitions are not proposed for inclusion in the PDTS since the terms are not used in any PDTS specification and do not apply to a facility in the permanently defueled condition. These terms are meaningful only to a reactor authorized to contain fuel and operate at power. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2).

Definitions Proposed for Deletion

1.25 FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2. All Surveillance Requirements shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. The 25% extension applies to all frequency intervals with the exception of "F." No extension is allowed for intervals designated "F."

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shiftly (once per 12 hours)
D	Daily (once per 24 hours)
W	Weekly (once per 7 days)
M	Monthly (once per 31 days)
Q	Quarterly (once per 92 days)
S/A	Semi Annually (once per 184 days)
R	Refueling Interval (once per 24 months)
P S/U	Prior to each reactor startup, if not done during the previous 7 days
P S/A	Within six (6) months prior to each reactor startup
P	Completed prior to each release
N/A (NA)	Not applicable
E	Once per 18 months
F	Not to exceed 24 months

Basis for Deletion
This definition is being reformatted, revised, and relocated to Section 3/4.0 "Limiting Conditions for Operation and Surveillance Requirement Applicability," as SR 4.0.3. The proposed change will ensure the appropriate requirements for the 25% grace period are maintained (see discussion of SR 4.0.3). The portion of the definition with respect to Frequency Notation and Table 1.2 are being deleted due to the elimination of most of the surveillance requirements. The wording of the proposed specification is from SR 3.0.2 in NUREG-1430 (Reference 6) and Draft NUREG-1625 (Reference 7), except that it is modified for a facility in permanently defueled condition.

TS SECTION 2 – SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	
Current TMI TS	**Proposed TMI TS**
TS 2.1 – Safety Limits – Reactor Core TS 2.2 – Safety Limits – Reactor System Pressure TS 2.3 – Limiting Safety System Settings, Protective Instrumentation	TS 2.1 – Deleted TS 2.2 – Deleted TS 2.3 – Deleted
BASIS	
TS Section 2, Safety Limits and Limiting Safety System Settings, contains "safety limits" and "limiting safety system settings" to establish limits on important process variables to assure the integrity of the fuel cladding and the RCS in all Modes of operation. Pursuant to 10 CFR 50.36(c)(1), safety limits are limiting parameters necessary to protect the physical barriers that guard against uncontrolled release of radioactivity from a nuclear reactor. The Safety Limits established in TS 2.1 and 2.2 protect the integrity of the fuel cladding and RCS barriers, respectively. Limiting safety system settings in TS 2.3 are values of various parameters associated with the Nuclear Steam Supply System at which automatic protective action is needed during normal operations or anticipated transients to prevent exceeding a safety limit. TS 2.1, Safety Limits – Reactor Core, provides safety limits that maintains the integrity of the fuel cladding and prevents fission product release. This specification applies to reactor thermal power, axial power imbalance, RCS pressure, coolant temperature, and coolant flow during power operation of the plant. TS 2.1 and Figures 2.1-1 and 2.1-3, "Reactor Outlet Temperature," are not proposed for inclusion in the PDTS. Pursuant to 10 CFR 50.82(a)(2), the facility license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor. Since the safety limits and limiting safety system settings apply to an operating reactor, they have no function in the permanently defueled condition. TS 2.2, Safety Limits – Reactor System Pressure, provides a maximum safety limit to maintain the integrity of the RCS and prevent the release of significant amounts of fission products activity. This specification is applicable during Power Operation and when fuel is in the reactor vessel. TS 2.2 is not proposed for inclusion in the PDTS. Pursuant to 10 CFR 50.82(a)(2), the facility license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor. Since the safety limits and limiting safety system settings apply to an operating reactor, they have no function in the permanently defueled condition. TS 2.3, Limiting Safety System Settings, Protective Instrumentation, provide limiting safety system settings to ensure the safety limits in TS 2.1 and 2.2 are not exceeded. The TS establishes the trip settings for automatic protective devices that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the facility during Normal Power Operation or Operational Transients conditions. TS 2.3 and Table 2.3-1, "Reactor Protection System Trip Setting Limits," are not proposed for	

inclusion in the PDTS. Pursuant to 10 CFR 50.82(a)(2), the facility license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor. Since the safety limits and limiting safety system settings apply to an operating reactor, they have no function in the permanently defueled condition.

Summary:

This section is being proposed for deletion in its entirety, since the safety limits do not apply to a reactor that is in a permanently defueled condition. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). These specifications do not apply to the safe storage and handling of spent fuel in the SFP. The actions for Safety Limit violations have been proposed for removal in TS LAR dated November 10, 2017 (Reference 9).

TS SECTION 3 – LIMITING CONDITION FOR OPERATION

TS Section 3 of the current TMI TS contains the Limiting Conditions for Operation (LCO). In accordance with 10 CFR 50.36(c)(2), LCOs specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The LCOs typically place restrictions on availability of safety equipment needed to prevent or mitigate a postulated Design Basis Accident (DBA), or on process variables necessary to preserve the initial conditions assumed in the safety analyses of postulated DBAs. 10 CFR 50.36(c)(2)(ii) defines four criteria for establishing LCOs (see Section 3.1). Associated surveillance requirements help to ensure that specified equipment and parameters are maintained within the limits specified in the LCOs.

As discussed previously, only postulated DBA (i.e., the Post Permanent Shutdown FHA) remains applicable relative to the TMI TS with the reactor in the permanently defueled state. As a result, this section is being revised to reflect only the limitations associated with remaining plant systems. The remaining systems are consistent with those required for a plant that has submitted certification that the reactor vessel will be maintained in a permanently defueled state. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), the sections that are no longer applicable are proposed to be deleted in their entirety and the revised sections are proposed to be changed to meet requirements that reflect the permanently defueled condition.

With the TS sections deleted or revised the applicable TS Bases sections will also be removed or changed.

Due to the reduced number of LCOs and Surveillance Requirements, TMI proposes to combine the LCOs (TS Section 3) with the corresponding Surveillance Requirements (TS Section 4). This format will allow the Surveillance Requirements to be more readily associated with the corresponding LCO. The section header for the Section 3/4 combined TS Section is proposed to be retitled "***3/4. LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS***." The LCOs and combined Surveillance Requirements (SR) sections will be designated with notation 3/4.#. The proposed format to the LCOs is shown in Attachment 2.

It is proposed to combine TS Section 3.0 and Section 4.0 for TS LCO General Action Requirements and Surveillance Standards into a common specification. TS Section 3.0, "GENERAL ACTION REQUIREMENTS," is proposed to be retitled "***3/4.0 GENERAL ACTION REQUIREMENTS AND SURVEILLANCE REQUIREMENT APPLICABILITY***." This change is editorial in nature. The proposed format for Section 3/4.0 is shown in Attachment 2.

It is proposed to renumber and retitle TS LCO 3.11, "Handling of Irradiated Fuel," as LCO "***3/4.1.4 HANDLING AND STORAGE OF IRRADIATED FUEL WITH THE FUEL HANDLING BUILDING CRANE***," and add new specifications to address operability requirements for the SFP Water Level, Boron

Concentration, and Spent Fuel Assembly Storage, and Fuel Handling Building Crane operation. This specification is discussed more in detail below.

The list below contains a comparison between the provisions of the current TMI TS and the proposed PDTS. Each subsection of TMI TS Section 3 is discussed in more detail in the tables below.

Current TMI LCO	Proposed PDTS
3.0 General Action Requirements	3/***4***.0 General Action Requirements ***and Surveillance Requirement Applicability***
3.1 Reactor Coolant System	Delete ***3/4.1 Handling and Storage of Irradiated Fuel in the Spent Fuel Pool***
3.2 Makeup and Purification and Chemical Addition Systems	Previously Deleted
3.3 Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems	Delete
3.4 Decay Heat Removal (DHR) Capability	Delete
3.5 Instrumentation Systems	Delete
3.6 Reactor Building	Delete
3.7 Unit Electric Power System	Delete
3.8 Fuel Loading and Refueling	Delete
3.9	Previously Deleted
3.10 Miscellaneous Radioactive Material Sources	Delete
3.11 Handling of Irradiated Fuel	Revised/Renumbered/Retitle as 3/4.1
3.12 Reactor Building Polar Crane	Delete
3.13 Secondary Coolant System Activity	Delete
3.14 Flood	Delete
3.15 Air Treatment Systems	Delete
3.16 Shock Suppressors (Snubbers)	Delete
3.17 Reactor Building Air Temperature	Delete
3.18	Previously Deleted
3.19 Containment Systems	Delete
3.20	Previously Deleted
3.21 Radioactive Effluent Instrumentation	Previously Deleted
3.22 Radioactive Effluents	Previously Deleted
3.23 Radiological Environmental Monitoring	Previously Deleted
3.24 Reactor Vessel Water Level Indication	Delete

<table>
<tr><th colspan="2">TS SECTION 3.0 – GENERAL ACTION REQUIREMENTS</th></tr>
<tr><td colspan="2">TS Section 3.0, General Action Requirements, contains the general requirements applicable to all LCOs and applies at all times unless otherwise stated in TS. Due to the limited number of LCOs in the proposed PDTS, a number of the TMI TS provisions in this section are no longer necessary or applicable to the TMI facility as indicated in the following table. LCOs 3.0.1 and 3.0.2 are being proposed for addition in the PDTS. These LCOs are based on NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants" (Reference 6) and Draft NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants" (Reference 7), which have been modified to reflect the permanently defueled condition.</td></tr>
<tr><th>Current TMI LCO</th><th>Basis for Deletion</th></tr>
<tr><td>LCO 3.0.1 When a Limiting Condition for Operation is not met, except as provided in action called for in the specification, within one hour action shall be initiated to place the unit in a condition in which the specification does not apply by placing it, as applicable, in:
1. At least HOT STANDBY within the next 6 hours.
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.
Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the time limits of the specification as measured from the time of failure to meet the Limiting Condition for Operation. Applicability of these requirements is stated in the individual specifications.
Specification 3.0.1 is not applicable in COLD SHUTDOWN OR REFUELING SHUTDOWN.</td><td>LCO 3.0.1 establishes limitations on changes in operational conditions or other specified conditions in an operating plant when an LCO is not met. It allows placing the operating plant in an operational condition or other specified condition stated in the Applicability when plant conditions are such that the requirements of the LCO would not be met.
This LCO and its Bases are not proposed for inclusion in the PDTS since LCO 3.0.1 will no longer be applicable. Pursuant to 10 CFR 50.82(a)(2), the facility license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor.</td></tr>
<tr><th colspan="2">Proposed TS Section 3/4.0 <u>LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENT APPLICABILITY</u></th></tr>
<tr><th>Proposed TMI LCO/SR</th><th>Basis for Addition</th></tr>
<tr><td>3.0.1 LCOs shall be met during the specified conditions in the TS, except as provided in 3.0.2.</td><td>LCO 3.0.1 is proposed as an addition to the PDTS. The specification establishes the applicability statement within each individual TS as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the applicability statement of each specification).
LCO 3.0.1 is being textually aligned with LCO 3.0.1 in NUREG-1430 (Reference 6) and Draft NUREG-</td></tr>
</table>

	1625 (Reference 7), except that it is modified for a facility in permanently defueled condition (i.e., removed references to MODES and LCOs that are not used).
3.0.2 Upon discovery of a failure to meet an LCO, the required actions of the associated Conditions shall be met. ***If the LCO is met or is no longer applicable prior to expiration of the specified completion time(s), completion of the required action(s) is not required, unless otherwise stated.***	LCO 3.0.2 is proposed as an addition to the PDTS. The specification establishes that upon discovery of a failure to meet an LCO, the associated action shall be met. The completion time of each required action for an action condition is applicable from the point in time that an action condition is entered. The required actions establish those remedial measures that must be taken within specified completion times when the requirement of an LCO are not met. LCO 3.0.2 is being textually aligned with LCO 3.0.2 in NUREG-1430 (Reference 6) and Draft NUREG-1625 (Reference 7), except that it is modified for a facility in permanently defueled condition (i.e., removed references to LCOs that are not used).
4.0.1 Surveillance requirements shall be met during the specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in 4.0.2.	SR 4.0.1 is relocated from current TS Section 4.0 "Surveillance Standards" to immediately follow the LCO statement in proposed PDTS Section 3/4.0. See current TS Section 4.0 for justification for proposed wording.
4.0.2 If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. ***If the surveillance is not performed within the delay period, the LCO must***	SR 4.0.2 is being relocated from current TS Section 4.0 "Surveillance Standards" in its entirety to immediately follow the LCO statement in proposed PDTS Section 3/4.0. See current TS Section 4.0 for justification for proposed wording.

immediately be declared not met, and the applicable condition(s) must be entered. ***When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.***	
4.0.3 The specified frequency for each SR is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.	SR 4.0.3 is being proposed for addition to the Surveillance Requirement Applicability section. This specification is based upon the TMI TS Definition for "Frequency Notation," which states, in part, "All Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval." SR 4.0.3 is textually aligned with the first paragraph of SR 3.0.2 in NUREG-1430 (Reference 6) and Draft NUREG-1625 (Reference 7), except that it is modified for a facility in permanently defueled condition (i.e., the remaining paragraphs were not included since the proposed PDTS do not use the "once" or "once per basis terminology, nor are any exceptions taken to this SR). There is no change in intent for this statement and the TMI TS definition. Both statements provide an allowance for extending the frequency for performance of a SR to 1.25 times the interval specified in the frequency to facilitate scheduling or unforeseen problems that may prevent performance during normal intervals.

TS SECTION 3.1 – REACTOR COOLANT SYSTEM	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.1.1 – Operational Components	LCO 3.1.1 – Deleted
LCO 3.1.2 – Pressurizer Heatup and Cooldown Limitations	LCO 3.1.2 – Deleted
LCO 3.1.3 – Minimum Conditions for Criticality	LCO 3.1.3 – Deleted
LCO 3.1.4 – Reactor Coolant System (RCS) Activity	LCO 3.1.4 – Deleted
LCO 3.1.5 – Chemistry	LCO 3.1.5 – Deleted
LCO 3.1.6 – Leakage	LCO 3.1.6 – Deleted
LCO 3.1.7 – Moderator Temperature Coefficient of Reactivity	LCO 3.1.7 – Deleted
LCO 3.1.8 – Single Loop Restrictions	LCO 3.1.8 – Deleted
LCO 3.1.9 – Low Power Physics Testing Restrictions	LCO 3.1.9 – Deleted
LCO 3.1.10 – Control Rod Operation	LCO 3.1.10 – Previously Deleted

LCO 3.1.11 – Reactor Internals Vent Valves	LCO 3.1.11 – Deleted
LCO 3.1.12 – Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)	LCO 3.1.12 – Deleted
LCO 3.1.13 – Reactor Coolant System Vents	LCO 3.1.13 – Deleted

BASIS

TS Section 3.1, Reactor Coolant System, contains LCOs to assure the operability of RCS. The LCOs are related to plant components and functions that ensure safe operation of the reactor and mitigate the effects of reactor related postulated DBA. These LCOs do not provide protection for the cladding of fuel stored in the SFPs.

LCO 3.1.1, Operational Components, applies to the operating status of RCS components to specify those limiting conditions which must be met to ensure safe reactor operations. The LCO specifies certain conditions of the RCS components that shall be in operation including reactor coolant loops and associated reactor coolant pumps, steam generator integrity, and pressurizer safety valves. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). None of the described process variables, design features, or operating restrictions are applicable with the plant in the permanently defueled state. LCO 3.1.1 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.2, Pressurizer Heatup and Cooldown Limitations, establishes the limits associated with maintaining the vessel pressure and temperature (P-T) limits including the limitation established with heatup and cooldown rates to prevent encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary. The RCS heatup and cooldown rate limits in this section provide a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR Part 50, Appendices G and H. The requirements of 10 CFR Part 50, Appendice G and H no longer apply because the reactor coolant pressure boundary will no longer be used as a fission product barrier when the reactor vessel is permanently defueled. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.2, including Figures 3.1-1, "Reactor Coolant System Heatup and Criticality Limitations"; 3.1-2, "Reactor Coolant System Cooldown Limitations"; and 3.1-3, "Reactor Coolant Inservice Leak Hydrostatic Test" are not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.3, Minimum Conditions for Criticality, applies to RCS conditions required prior to criticality. It ensures certain parameters are maintained for safe power operations. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.3 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.4, Reactor Coolant System (RCS) Activity, establishes limits for RCS Dose Equivalent I-131 and Dose Equivalent Xe-133 specific activity commensurate with the offsite and control room doses. This specification provides protection for the Steam Line Break and Steam Generator Tube Rupture events to ensure that the dose would be within acceptable limits. RCS Activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). With the reactor vessel in a permanently defueled state, the applicable UFSAR Chapter 14 postulated accidents are no longer credible. LCO 3.1.4 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.5, Chemistry, establishes limits during reactor operation to protect the RCS materials from stress corrosion cracking. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.5 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.6, Leakage, establishes process variable limits and operating restrictions for RCS pressure boundary leakage, unidentified RCS leakage, identified RCS leakage, and primary to secondary leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event. With the reactor vessel in a permanently defueled state, the applicable UFSAR Chapter 14 postulated accidents are no longer credible. The RCS Operational Leakage LCO satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.6 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.7, Moderator Temperature Coefficient of Reactivity, defines the safe operating limits for reactor moderator temperature coefficient (MTC) for the reactor core during full power conditions. This LCO for MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.7 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.8, Single Loop Restrictions, defines the prohibition of single loop operations of the RCS when the reactor is critical. This LCO satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.8 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.9, Low Power Physics Testing Restrictions, defines additional conditions for Low Power Physics Testing (LPPT) to assure an additional margin of safety during LPPT. This specification satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.9 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.10, Control Rod Operation, was previously deleted and is not proposed for inclusion in the PDTS.

LCO 3.1.11, Reactor Internals Vent Valves, verifies the reactor internals vent valves exhibit freedom of movement so that they will maintain their design function during a Large Break Loss of Coolant Accident (LOCA). This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.10 is not proposed for inclusion in the PDTS since a Large Break LOCA will not be a credible event since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.12, Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP), defines conditions for LTOP protection to prevent the possibility of inadvertently over-pressurizing or depressurizing the RCS. This specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.12 is not proposed for inclusion in the PDTS since the RCS will no longer require LTOP protection since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.1.13, Reactor Coolant System Vents, defines the conditions for operation of the RCS vents to ensure that sufficient vent flow paths are operable when the reactor is critical. RCS vents ensures capability of venting non-condensible gases from the RCS. The basis is to ensure a method and system is available to remove steam and/or non-condensible gases from the RCS, which may inhibit core cooling during natural circulation. This specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). LCO 3.1.13 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

<u>Summary</u>:

The current content of this section is being proposed for deletion in its entirety. All of these LCOs are related to assuring the integrity of the RCS pressure boundary for an operating reactor. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the RCS components and functions addressed in TS

Section 3.1 will no longer apply or be required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3/4.1 – <u>HANDLING AND STORAGE OF IRRADIATED FUEL IN THE SPENT FUEL POOL</u>

3/4.1.1 SPENT FUEL POOL WATER LEVEL

Applicability

Applies to the minimum level of water in the Spent Fuel Pool during handling of irradiated fuel in the Spent Fuel Pool.

Objective

Ensures that assumptions of Fuel Handling Accident are maintained during handling of irradiated fuel in the Spent Fuel Pool.

Specification

3.1.1.1 Maintain Spent Fuel Pool level greater than or equal to 342’4” elevation.

3.1.1.2 With Spent Fuel Pool level less than 342’4” elevation, immediately suspend handling of irradiated fuel in the Spent Fuel Pool.

SURVEILLANCE REQUIREMENTS

4.1.1.1 Verify Spent Fuel Pool level greater than or equal to 342’4” elevation every 7 days.

BASIS

A new LCO is proposed that will specify a minimum water level as expressed in spent fuel pool elevation that will be applicable during fuel handling activities. The top of fuel is at the 319’4” elevation. The Post Permanent Shutdown FHA analysis assumes 23 feet of water above the fuel assemblies. This dictates a minimum elevation of water in the SFP of 342’4”. This specification provides the controls to ensure the assumptions of the accident analysis while fuel handling evolutions are in progress, and satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). This specification will have a Surveillance Requirement SR 4.1.1.1 that will verify the SFP water level on a frequency of 7 days.

The water contained in the SFP provides a medium for removal of decay heat from the stored fuel elements, normally via the spent fuel cooling system. The SFP water also provides shielding to reduce the general area radiation dose during both spent fuel handling and storage. The resultant 2-hour dose to a person at the exclusion area boundary and the 30-day dose at the low population zone are much less than 10 CFR 50.67 limits.

LCO 3.1.1.2 requires that when the water level in the SFP is lower than the required level, the movement of irradiated fuel assemblies in the SFP is to be "immediately" suspended. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, such that the suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring in the SFP when the level is below the required elevation.

Although maintaining adequate SFP water level is essential to both decay heat removal and shielding effectiveness, the Technical Specification minimum water level limit is based upon maintaining the pool's iodine retention-effectiveness consistent with that assumed in the evaluation of the Post Permanent Shutdown FHA analysis. The Post Permanent Shutdown FHA analysis assumes that a minimum of 23 feet of water is maintained above the stored fuel. This assumption allows the use of the pool iodine decontamination factor of 200 used in the associated offsite dose calculation.

3/4.1.2 SPENT FUEL POOL BORON CONCENTRATION

Applicability

Applies to the minimum boron concentration in the Spent Fuel Pool during storage and handling of irradiated fuel in the Spent Fuel Pool.

Objective

Ensures that assumptions of Storage Limitations are maintained to prevent inadvertent criticality in the spent fuel pool.

Specification

3.1.2.1 Maintain Spent Fuel Pool boron concentration greater than or equal to 600 ppm.

3.1.2.2 With Spent Fuel Pool boron concentration less than 600 ppm, immediately suspend handling of irradiated fuel in the Spent Fuel Pool and immediately restore boron concentration per 3.1.2.1.

SURVEILLANCE REQUIREMENTS

4.1.2.1 Verify Spent Fuel Pool boron concentration greater than or equal to 600 ppm every 7 days.

BASIS

A new LCO is proposed that will specify a minimum boron concentration of 600 ppm that will be applicable anytime irradiated fuel is stored in the spent fuel pool. This specification is relocated from current design feature 5.4.1.a. This specification will have a Surveillance Requirement SR 4.1.2.1 that will verify the SFP boron on a frequency of 7 days. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO 3.1.2.2 requires that when the SFP boron concentration is less than 600 ppm, the movement of irradiated fuel assemblies in the SFP is to be "immediately" suspended. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, such that the suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring in the SFP when the boron concentration is below the required level.

The acceptance criteria for the fuel storage pool criticality analyses is that a keff of < 0.95 must be maintained for all postulated events. The storage racks are capable of maintaining this keff with unborated pool water at a temperature yielding the highest reactivity (assuming the storage restrictions of LCO 3.1.3 are met). Most abnormal storage locations will not result in an increase in the keff of the racks. However, it is possible to postulate events, such as the mis-loading of an assembly with a burnup and enrichment combination outside the acceptable area in Figure 3.1.3-1 and 3.1.3-2, or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. For such events, credit is taken for the presence of boron in the pool water since the NRC does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in keff, caused by the boron more than offsets the reactivity addition caused by credible accidents.

3/4.1.3 SPENT FUEL ASSEMBLY STORAGE

Applicability

Applies whenever any fuel assembly is stored in Storage Pool A or Storage Pool B of the Spent Fuel Pool.

Objective

Ensures that assumptions of Storage Limitations are maintained to prevent inadvertent criticality in the Spent Fuel Pool.

Specification

3.1.3.1 The combination of initial enrichment and burnup of each spent fuel assembly stored in Storage Pool A and Storage Pool B, shall be within the acceptable region of Figure 3.1.3-1 or 3.1.3-2.

3.1.3.2 When requirement of 3.1.3.1 is not met, immediately initiate action to move the noncomplying fuel assembly to an acceptable configuration.

SURVEILLANCE REQUIREMENTS

4.1.3.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.1.3-1 or Figure 3.1.3-2 prior to storing irradiated spent fuel in the Spent Fuel Pool A or Spent Fuel Pool B.

BASIS

This LCO is proposed to specify the permissible combined initial enrichment and burnup limits of fuel to be Stored in Spent Fuel Pool A Region II and Spent Fuel Pool B. The limits are relocated from TS 5.4 to this new proposed LCO. This specification will have a Surveillance Requirement SR 4.1.3.1 that will verify the loading requirements. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO 3.1.3.2 requires that when LCO 3.1.3.1 is not met, "immediately" initiate action to move the noncomplying fuel assembly to an acceptable configuration. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, to reestablish the safety margins to prevent an inadvertent criticality.

The function of the spent fuel storage racks is to support safety analyses and protect spent fuel assemblies from the time they are placed in the pool until they are shipped offsite. The spent fuel assembly storage LCO was derived from the need to establish limiting conditions on fuel storage to assure sufficient safety margin exists to prevent inadvertent criticality. The spent fuel assemblies are stored entirely underwater in a configuration that has been shown to result in a reactivity of less than or equal to 0.95 under worst case conditions. The spent fuel assembly enrichment requirements in this LCO are required to ensure inadvertent criticality does not occur in the spent fuel pool. Inadvertent criticality within the fuel storage area could result in offsite radiation doses exceeding 10 CFR 50.67 limits.

3/4.1.4 HANDLING OF IRRADIATED FUEL WITH THE FUEL HANDLING BUILDING CRANE

TS LCO 3.11 is being proposed to be relocated to LCO 3/4.1.4. See LCO 3.11 for description.

BASIS

The proposed changes are described in LCO 3.11; this LCO will be retained with minor editorial changes.

TS SECTION 3.2 – MAKEUP AND PURIFICATION AND CHEMICAL ADDITION SYSTEM

TS Section 3.2 "Makeup and Purification and Chemical Addition System," was previously deleted.

TS SECTION 3.3 – EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Current TMI LCO	Proposed TMI LCO
LCO 3.3.1.1 – Injection Systems	LCO 3.3.1.1 – Deleted
LCO 3.3.1.2 – Core Flooding System	LCO 3.3.1.2 – Deleted
LCO 3.3.1.3 – Reactor Building Spray System and Reactor Building Emergency Cooling System	LCO 3.3.1.3 – Deleted
LCO 3.3.1.4 – Cooling Water Systems	LCO 3.3.1.4 – Deleted
LCO 3.3.1.5 – Engineered Safeguards Valves and Interlocks	LCO 3.3.1.5 – Deleted
LCO 3.3.2 -3.3.4 – Maintenance requirements during operation	LCO 3.3.2 -3.3.4 – Deleted

BASIS

TS Section 3.3, Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems, contains LCOs to assure the operability of the emergency cooling systems and to provide assurance of adequate cooling capability for heat removal in the event of a LOCA or isolation from the normal reactor heat sink.

10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,* specifies that light-water nuclear power reactors must have Emergency Core Cooling System (ECCS) designed to meet the cooling performance requirements following postulated LOCAs. However, 10 CFR 50.46(a)(1)(i) states *"This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted."*

LCO 3.3.1.1, Injection Systems, defines the ECCS injection systems required to be in service prior to the reactor being placed in a critical condition. This required a high pressure injection system, low pressure injection system with an associated borated water source. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.3.1.1 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.3.1.2, Core Flooding System, defines the operability requirement for the Core Flooding Tanks. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.3.1.2 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.3.1.3, Reactor Building Spray System and Reactor Building Emergency Cooling System, defines the operability requirements for containment pressure control system required to maintain containment conditions following a LOCA. In addition, this specification provides for Reactor Building emergency sump pH control using trisodium phosphate dodecahydrate (TSP). This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.3.1.3 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.3.1.4, Cooling Water Systems, defines the operability of the cooling support systems, necessary to remove heat from the RCS in the event of a LOCA. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.3.1.4 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.3.1.5, Engineered Safeguards Valves and Interlocks, defines the operability of the valves and interlocks required to support the system defined in TS 3.3.1.1 through 4. This specification satisfies Criterion

3 of 10 CFR 50.36(c)(2)(ii). LCO 3.3.1.5 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.3.2 through 3.3.4, Maintenance requirements during operation, defines the conditions that must be met to remove a system required by 3.3.1 from service to perform maintenance on the ECCS system when the reactor is critical. LCOs 3.3.2 through 3.3.4 are not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

Summary:

The section is being proposed for deletion in its entirety. These LCOs are related to providing cooling for a reactor core and maintaining containment integrity in the event of a LOCA pursuant to 10 CFR 50.46. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the core and reactor building cooling and spray system specifications addressed in TS Section 3.3 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.4 – DECAY HEAT REMOVAL (DHR) CAPABILITY	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.4.1 – Reactor Coolant System (RCS) temperature greater than 250 degrees F LCO 3.4.2 – RCS temperature less than or equal to 250 degrees F	LCO 3.4.1 – Deleted LCO 3.4.2 – Deleted

BASIS

TS Section 3.4, Decay Heat Removal (DHR) Capability, defines the conditions necessary to assure the operability of the systems designed to remove decay heat when one or more fuel assemblies are located in the reactor pressure vessel (RPV). Normal DHR is by the Once Thru Steam Generators (OTSG) with the steam dump to the condenser when RCS temperature is above 250 degrees F and by the DHR System below 250 degrees F. Normally, the capability to return feedwater flow to the OTSGs is provided by the main feedwater system.

LCO 3.4.1, Reactor Coolant System (RCS) temperature greater than 250 degrees F, establishes the requirement to have emergency feedwater (EFW) pumps, condensate storage tank, turbine bypass valves, and main steam safety valves operable. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.4.1 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.4.2, RCS temperature less than or equal to 250 degrees F, establishes the requirements for the DHR system. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). The DHR system is not required when there is no fuel in the RPV. LCO 3.4.2 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

Summary:

The section is being proposed for deletion in its entirety. The DHR capability is relied upon for decay heat removal from the RCS. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the RCS decay heat cooling specifications addressed in TS Section 3.4 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.5 – INSTRUMENTATION SYSTEM

Current TMI LCO	Proposed TMI LCO
LCO 3.5.1 – Operational Safety Instrumentation	LCO 3.5.1 – Deleted
LCO 3.5.2 – Control Rod Group and Power Distribution Limits	LCO 3.5.2 – Deleted
LCO 3.5.3 – Engineered Safeguards Protection System Actuation Setpoint	LCO 3.5.3 – Deleted
LCO 3.5.4 – Incore Instrumentation	LCO 3.5.4 – Deleted
LCO 3.5.5 – Accident Monitoring Instrumentation	LCO 3.5.5 – Deleted
LCO 3.5.6 – Previously Deleted	LCO 3.5.6 – Deleted
LCO 3.5.7 – Remote Shutdown System	LCO 3.5.7 – Deleted

BASIS

TS Section 3.5, Instrumentation System, contains LCOs to assure the operability of protective instrumentation. The LCOs are related to plant instrumentation that performs protective and monitoring functions to ensure safe operation of the reactor and mitigate the effects of reactor-related postulated DBAs.

LCO 3.5.1, Operational Safety Instrumentation, provides the operability requirements for the Reactor Protection System, Engineered Safety Features, and Heat Sink Protection System specified in Table 3.5-1, Instruments Operating Conditions. Table 3.5-1 defines, for each protective function, the minimum number of operable channels and the minimum degree of redundancy, and the specified operator action if the previously stated conditions cannot be met. The objective of the Operational Safety Instrumentation is to delineate conditions of the unit instrumentation and safety circuits necessary to assure reactor safety. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). With the reactor permanently defueled, no condition will exist in which the applicable UFSAR Chapter 14 postulated accidents are credible. LCO 3.5.1, including Table 3.5-1, is not proposed for inclusion in the PDTS since there will no longer be a need for Operational Safety Instrumentation to protect the reactor core since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.5.2, Control Rod Group and Power Distribution Limits, includes specifications for axial power imbalance, quadrant power tilt (QPT), and control rod position limit. The specification assures an acceptable core power distribution during power operation, sets a limit on potential reactivity insertion from a hypothetical control rod ejection, and assures core subcriticality after a reactor trip. These limits are based on the LOCA analysis which defines the maximum linear heat rate. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). With the reactor permanently defueled, no condition will exist in which the applicable UFSAR Chapter 14 postulated accidents are credible. LCO 3.5.2 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.5.3, Engineered Safeguards Protection System Actuation Setpoint, establishes requirements for the engineered safeguards protection system actuation setpoints to initiate the necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). With the reactor permanently defueled, no condition will exist in which the applicable UFSAR Chapter 14 postulated accidents are credible. LCO 3.5.3 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.5.4, Incore Instrumentation, was previously deleted. Figure 3.5-1, Incore Instrumentation Specification Axial Imbalance Indication, Figure 3.5-2, Incore Instrumentation Specification Radial Flux Indication, and Figure 3.5-3, Incore Instrumentation Specification; aid in determining the operability of the full incore system for measurement of QPT and axial power imbalance. specified in LCO 3.5.2 above. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). The Figures, including Figures 3.5-1, 3.5-2, and 3.5-3 are not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.5.5, Accident Monitoring Instrumentation, establishes requirements for accident monitoring and post-accident monitoring instrumentation specified in Table 3.5-2, Accident Monitoring Instruments, and Table 3.5-3, Post Accident Monitoring Instrumentation, respectively. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). With the reactor permanently defueled, no condition will exist in which the applicable UFSAR Chapter 14 postulated accidents are credible. LCO 3.5.5 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.5.6 was previously deleted.

LCO 3.5.7, Remote Shutdown System, identifies the Remote Shutdown functions that must be operable as shown in Table 3.5-4, Remote Shutdown System Instrumentation and Controls. The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the reactor in a safe hot shutdown condition from locations other than the control room. This specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). LCO 3.5.7 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

Summary:

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications for the instrumentation system addressed in TS Section 3.5 are not required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.6 – REACTOR BUILDING	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.6.1 – Conditions that require Containment Integrity (CI)	LCO 3.6.1 – Deleted
LCO 3.6.2 – Conditions that require CI with RCS open	LCO 3.6.2 – Deleted
LCO 3.6.3 – Positive Reactivity Insertions	LCO 3.6.3 – Deleted
LCO 3.6.4 – Reactor building internal pressure limits	LCO 3.6.4 – Deleted
LCO 3.6.5 – Containment Isolation Valves (CIVs) positions	LCO 3.6.5 – Deleted
LCO 3.6.6 – Containment Isolation Valves (CIVs) inoperable	LCO 3.6.6 – Deleted

LCO 3.6.7 – Previously Deleted	LCO 3.6.7 – Deleted
LCO 3.6.8 – 48" reactor building purge valve inoperable	LCO 3.6.8 – Deleted
LCO 3.6.9 – Previously Deleted	LCO 3.6.9 – Deleted
LCO 3.6.10 – During Startup, Hot Standby and Power Operation Conditions	LCO 3.6.10 – Deleted
LCO 3.6.11 – Reactor in Cold Shutdown or Refueling Shutdown conditions	LCO 3.6.11 – Deleted
LCO 3.6.12 – Personnel or emergency air locks	LCO 3.6.12 – Deleted

BASIS

TS Section 3.6, Reactor Building, established the requirements that assure containment integrity. The containment, including all its penetrations, is designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. The containment and internal structures also provides shielding from the fission products that may be present in the containment atmosphere following accident conditions. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO 3.6.1 specifies that containment integrity required whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.
b. Reactor coolant temperature is 200 degrees F or greater.
c. Nuclear fuel is in the core.

LCO 3.6.2 specifies that containment integrity exists whenever the RCS is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn.

LCO 3.6.3 restricts positive reactivity insertions unless containment integrity is maintained.

LCO 3.6.4 requires reactor shutdown when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

LCO 3.6.5 requires a check shall be made to confirm that all manual CIVs which should be closed are closed and are conspicuously marked.

LCO 3.6.6 specifies actions if a CIV (other than a purge valve) is determined to be inoperable.

LCO 3.6.8 specifies actions if a 48" Reactor Building purge valve is determined to be inoperable.

LCO 3.6.9 specifies limiting operation of Reactor Building purge isolation valves.

LCO 3.6.10 specifies containment integrity during Startup, Hot Standby, and Power Operation.

LCO 3.6.11 specifies containment integrity during Cold Shutdown or Refueling Shutdown.

LCO 3.6.12 specifies requirements for the personnel or emergency air locks.

LCO 3.6.1 through LCO 3.6.12 listed above not proposed for inclusion in the PDTS, since all fuel will be permanently removed from the RPV and stored in the SFP located in the Fuel Handling Building. With the reactor permanently defueled, no condition will exist in which the applicable UFSAR Chapter 14 postulated accidents are credible. Therefore, the requirements for the Reactor Building are no longer necessary or applicable to protect the health and safety of the public.

Summary:

The TS section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications for the Reactor Building addressed in TS Section 3.6 are not required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.7 – UNIT ELECTRIC POWER SYSTEM	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.7.1 – Defines minimum Electrical Power Systems requirements to place the reactor in a critical state.	LCO 3.7.1 – Deleted
LCO 3.7.2 – Defines the Allowable Outage Times for Electrical Power Systems while the reactor is critical.	LCO 3.7.2 – Deleted

BASIS

TS Section 3.7, Unit Electric Power System, contains LCOs related to the operability of AC and DC electrical systems. This section establishes the requirements for appropriate functional capability of plant electrical equipment required for safe operation of the facility. This section specifies requirements to ensure that the station safety-related electrical bussing and distribution system, offsite power sources, and the onsite standby power sources (emergency diesel generators (EDG)), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. The requirements for the EDG fuel oil storage are included for each EDG. Also included in this section is the requirements for direct current (DC) power. It specifies requirements to ensure that the DC electrical power subsystems are operable. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO 3.7.1 describes the minimum electrical power system alignment. This includes the following:

a. All engineered safeguard buses, engineered safeguards switchgear, and engineered safeguards load shedding systems are operable;
b. One 7200 volt bus is energized;
c. Two 230 KV lines are in service;
d. One 230 KV bus is in service;
e. Engineered safeguards diesel generators are operable and at least 25,000 gallons of fuel oil are available in the storage tanks;
f. Station batteries are charged and in service. Two battery chargers per battery are in service.

These systems ensure that the systems assumed to be operable to function during a design basis accident analysis are available.

LCO 3.7.1 is not proposed for inclusion in the PDTS. The design basis accidents and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA in the SFP. Exelon performed a calculation (Reference 10) for a FHA in the SFP that shows the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event (after 60 days of irradiated fuel decay time after reactor shutdown and compliance with the SFP water level requirements in proposed TS 3/4.1).

During movement of irradiated fuel assemblies in the SFP, there are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA with the unit permanently defueled. Because the FHA analysis does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the alternating current (AC) sources are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. As such, the requirement for AC and DC sources are being deleted because there are no design basis events that rely on these sources for mitigation.

LCO 3.7.2 provides the allowable outage times for equipment specified in LCO 3.7.1 in the event equipment becomes unavailable and dictates the actions necessary to ensure the plant is placed in an operational condition based on required equipment status. Since the equipment is not required to support the station in a permanently shutdown and defueled condition, based on accident analysis necessary to support this plant condition, and is proposed to be deleted, there will be no need for allowable outage times. Therefore, the conditions of LCO 3.7.2 are not proposed for inclusion in the PDTS.

Summary:

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.7 are not required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.8 – FUEL LOADING AND REFUELING	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.8.1 – Previously Deleted	LCO 3.8.1 – Deleted
LCO 3.8.2 – Core subcritical neutron monitors	LCO 3.8.2 – Deleted
LCO 3.8.3 – Decay heat removal pump and cooler	LCO 3.8.3 – Deleted
LCO 3.8.4 – Boron Concentration	LCO 3.8.4 – Deleted
LCO 3.8.5 – Direct communications Reactor Building to control room	LCO 3.8.5 – Deleted
LCO 3.8.6 – Reactor Building air-lock doors	LCO 3.8.6 – Deleted
LCO 3.8.7 – Reactor Building penetrations during fuel moves	LCO 3.8.7 – Deleted
LCO 3.8.8 – Conditions to stop fuel movement in the core	LCO 3.8.8 – Deleted
LCO 3.8.9 – Associated radiation monitors	LCO 3.8.9 – Deleted
LCO 3.8.10 – No irradiated fuel removed until 72 hours subcritical	LCO 3.8.10 – Deleted
LCO 3.8.11 – Maintain 23 feet of water above RPV flange	LCO 3.8.11 – Deleted

BASIS
TS Section 3.8, Fuel Loading and Refueling, establishes the specifications for refueling and fuel loading into the RPV in the Reactor Building. These LCOs are applicable when irradiated fuel is located within the RPV and do not apply to the safe storage and handling of spent fuel in the SFP. The LCO specifies neutron monitoring, boron concentration, and core cooling during refueling operations; requirements for containment airlocks and reactor building penetrations, radiation monitoring, minimum water level above the RPV flange, and maintaining direct communications between the Reactor Building and the main control room. These specifications satisfy Criterion 2 and Criterion 3 of 10 CFR 50.36(c)(2)(ii). LCO 3.8.1 through LCO 3.8.11 listed above are not proposed for inclusion in the PDTS, since all refueling operations will be completed and all fuel will be permanently removed from the RPV and stored in the SFP located in the Fuel Handling Building (or the ISFSI once constructed). Fuel loading and refueling activities will not occur at TMI since the TMI license will no longer be authorized to place or retain fuel in the reactor vessel in a permanently defueled condition. Therefore, the requirements for Fuel Loading and Refueling are no longer applicable. <u>Summary</u>: The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.8 are not required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.9 – DELETED
TS Section 3.9 was previously removed. The remaining reference to TS 3.9 is proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 3.10 – MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.10.1.1 – The source leakage test performed pursuant to Specification 4.13 <…>	LCO 3.4.1 – Deleted
LCO 3.10.1.2 – A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.	LCO 3.4.2 – Deleted
BASIS	
TS Section 3.10, Miscellaneous Radioactive Materials Sources, contains the specifications to assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation ensures that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. LCO 3.10.1.1 applies to each licensed sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting materials or 5 microcuries of alpha emitting material. This	

requirement ensures that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from a source material. This specification is not proposed for inclusion in the PDTS. This requirement is not credited in any safety analysis and does not meet any of the four screening criteria in 10 CFR 50.36(c)(2)(ii). Therefore, this specification can be deleted from the PDTS. Licensed sealed sources are controlled under a licensee controlled program.

LCO 3.10.1.2 requires a complete inventory of licensed radioactive materials in possession shall be maintained current at all times. This specification is not proposed for inclusion in the PDTS. This requirement is not credited in any safety analysis and does not meet any of the four screening criteria in 10 CFR 50.36(c)(2)(ii). Therefore, this specification can be deleted from the PDTS. Licensed sealed sources are controlled under a licensee controlled program.

Summary:

The section is being proposed for deletion in its entirety. These LCO requirements are not credited in any safety analysis and they do not meet any of the four screening criteria in 10 CFR 50.36(c)(2)(ii). Further, this TS Section is not included in the Standardized TS provided in NUREG-1430 (Reference 6) or Draft NUREG-1625 (Reference 7). The LCO does not satisfy the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.10 are not required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.11 HANDLING OF IRRADIATED FUEL	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.11 – Handling of Irradiated Fuel	***LCO 3.1.4–*** Handling of Irradiated Fuel ***with the Fuel Handling Building Crane***
Basis	
TS LCO Section 3.11 is proposed to be relocated, renumbered, and retitled as "***LCO 3.1.4 Handling of Irradiated Fuel with the Fuel Handling Building Crane***." This existing LCO’s title is changed to reflect the content of the specification is the handling of fuel using the Fuel Handling Building Crane vice the Spent Fuel Pool Fuel Handling Equipment. In addition, the number of the specification is changed as reflected in Attachment 2. The content of the LCO will remain unchanged. These changes are deemed editorial in nature.	

TS SECTION 3.12 – REACTOR BUILDING POLAR CRANE	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.12.1 – Reactor Building Crane operation during movement of fuel assemblies.	LCO 3.12.1 – Deleted
LCO 3.12.2 – During the period when the RPV head is removed.	LCO 3.12.2 – Deleted
LCO 3.12.3 – During the period when the RCS is pressurized	LCO 3.12.3 – Deleted

BASIS

TS Section 3.12, Reactor Building Polar Crane, contains the LCOs related to conditions for which the operation of the Reactor Building polar crane hoists are restricted. This TS Section applies to when the Reactor Building polar crane hoists is in use over the steam generator compartments and the fuel transfer canal.

LCO 3.12.1 and LCO 3.12.2 restricts the use of the reactor building polar crane hoists over the fuel transfer canal, when the reactor vessel head is removed, to preclude the dropping of materials or equipment into the reactor vessel and possibly damaging the fuel to the extent that any escape of fission products would result. These specifications satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). LCO 3.12.1 and LCO 3.12.2 are not proposed for inclusion in the PDTS since the TMI license will no longer be authorized to place or retain fuel in the reactor vessel in a permanently defueled condition.

LCO 3.12.3 restricts the use of the reactor building polar crane hoists over the steam generator compartments during the time when steam could be formed to prevent dropping a load on the steam generator or reactor coolant piping resulting in rupture of the system is required to protect against a LOCA. These specifications satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii). LCO 3.12.3 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized to place or retain fuel in the reactor vessel in a permanently defueled condition.

Summary:

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.12 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.13 – SECONDARY COOLANT SYSTEM ACTIVITY	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.13– Secondary Coolant System Activity	LCO 3.13 – Deleted

BASIS
TS Section 3.13, Secondary Coolant System Activity, limits secondary system specific activity as expressed as Dose Equivalent I-131 to ensure that resultant offsite radiation dose will be limited to a small fraction of the 10 CFR Part 100 limits in the event of a steam line rupture. The specification applies when RCS pressure is greater than 300 psig or T_{avg} is greater than 200°F. This specification satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). With the permanent cessation of power operations and permanently defuel status the Main Steam Line Break accident analysis is no longer applicable, and the plant conditions stated in the applicability and the applicable accident analysis supporting the permanently shutdown and defueled condition does not place any constraints on secondary coolant system activity. Summary: The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.13 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.14 – FLOOD	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.14.1 – Periodic Inspection of the Dikes Around TMI LCO 3.14.2 – Flood Condition for Placing the Unit in Hot Standby	LCO 3.14.1 – Deleted LCO 3.14.2 – Deleted
BASIS	
TS Section 3.14, Flood, contains LCOs related to flood protection. The design flood described in the TMI license basis is a Susquehanna River peak flow of 1,100,000 cubic feet per second (CFS). This event produces a peak water level of 301.6' elevation. The TMI site is elevated above this height and is surrounded by an earthen barrier (i.e., dike) which would prevent inundation of the site for river levels up to 304' elevation. Due to a change in the Susquehanna River probable maximum flood (PMF) during the original licensing process, TMI committed to provide for a safe and orderly shutdown for the revised PMF (LCO 3.14.2). The PMF is an event with a Susquehanna River peak flow of 1,625,000 CFS, a warning time of at least 30 hours, a peak river water level of 313.3' elevation, and a period of inundation of 50 hours. LCO 3.14.1, Periodic Inspection of the Dikes Around TMI, establishes the minimum frequency for inspection of the dikes and to define the flood stage after which the dikes will be inspected. LCO 3.14.1 is being not proposed for inclusion in the PDTS because this specification does not meet any of the four screening criteria in 10 CFR 50.36(c)(2)(ii). The dike is not a detector or indicator of reactor coolant pressure boundary degradation (Criterion 1); it is not a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis (Criterion 2); and is it not part of the primary success path in the mitigation of a postulated DBA or transient (Criterion 3); and after the reactor is permanently shut down and defueled, the risk of a radiological release significant to public health and safety is very low (Criterion 4). Since this LCO does not meet any of the four criteria, it may be removed from the PDTS.	

LCO 3.14.2, Flood Condition for Placing the Unit in Hot Standby, establishes requirements to place the reactor in hot standby when the river stage reaches 302' elevation. Specifically, LCO 3.14.2 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition, and as such, the requisite action to place the unit in a hot standby condition is no longer required.

Summary:

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.14 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.15 – AIR TREATMENT SYSTEMS	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.15.1 – Emergency Control Room Air Treatment System LCO 3.15.2 – Previously Deleted LCO 3.15.3 – Previously Deleted LCO 3.15.4 – Fuel Handling Building ESF Air Treatment System	LCO 3.15.1 – Deleted LCO 3.15.2 – Deleted LCO 3.15.3 – Deleted LCO 3.15.4 – Deleted
BASIS	

TS Section 3.15, Air Treatment Systems, contains requirements for the Emergency Control Room Air Treatment System, the Control Room Envelope (CRE) boundary, and the Fuel Handling Building ESF Air Treatment System.

LCO 3.15.1, Emergency Control Room Air Treatment System, establishes the requirements for the two independent systems that control the control room atmosphere for air intake and for recirculation within the CRE boundary. High efficiency particulate air (HEPA) filters and charcoal absorbers reduce the potential intake of radioiodine to the control room and maintain the dose less than the allowable levels for Control Room Habitability as stated in Criterion 19 of the General Design Criteria, Appendix A to 10 CFR Part 50. The Emergency Control Room Air Treatment System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The Emergency Control Room Air Treatment System is required to be operable at all times when containment integrity is required (e.g. RCS pressure greater than 300 psig, RCS Temperature is greater than 200F, and nuclear fuel is in the core) and/or irradiated fuel handling operations are in progress.

Following the permanent cessation of power operations, the DBAs associated with operations will no longer be applicable. The UFSAR Chapter 14 postulated DBA that remains applicable relative to TMI TS in the permanently shutdown and defueled condition is a FHA in the SFP. The Post Permanent Shutdown FHA analysis (Reference 10) concluded that the dose consequences are acceptable without relying on any SSCs to remain functional following 60 days of irradiated fuel decay time after reactor shutdown and compliance with the SFP water level requirements of proposed TS 3.1.1. However, Exelon proposes to prohibit movement of spent fuel after the submittal of the certification of permanent removal of fuel from the reactor vessel until 60 days after permanent shutdown through the imposition of the proposed License Condition. This will effectively prevent a FHA from occurring until after the 60-day decay period has elapsed and allows this LCO to be eliminated during the decay period. Therefore, the Emergency Control Room Air Treatment System and CRE are no longer required since they are no longer credited to protect the control room staff.

LCO 3.15.2 – Previously Deleted

LCO 3.15.3 – Previously Deleted

LCO 3.15.4, Fuel Handling Building ESF Air Treatment System, establishes the requirements for Fuel Handling Building ventilation during fuel movement and when its surveillance requirements are met. As discussed in the Fuel Handling Accident Analysis for the Permanently Defueled Condition section of this attachment, in the Post Permanent Shutdown FHA analysis there are no active systems credited as part of the initial conditions of the analysis or as part of the primary success path for mitigation of the FHA with the unit permanently defueled. Therefore, the use of the ESF Air Treatment is not credited or required in the FHA for reduction of nuclides or a reduction of onsite or offsite doses after 60 days of decay time. However, Exelon proposes to prohibit movement of spent fuel after the submittal of the certification of permanent removal of fuel from the reactor vessel until 60 days after permanent shutdown through the imposition of the proposed License Condition. This will effectively prevent a FHA from occurring until after the 60-day decay period has elapsed and allows this LCO to be eliminated during the decay period. LCO 3.15.4 is not proposed for inclusion in the PDTS.

Summary:

The section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the specifications addressed in TS Section 3.15 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.16 – SHOCK SUPPRESSORS (SNUBBERS)	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.16.1 – Safety-Related Snubber Operability	LCO 3.16.1 – Deleted
BASIS	

TS Section 3.16, Shock Suppressors (Snubbers), contains conditions to assure the operability of safety-related snubbers and establishes the actions that must be implemented when the LCO is not met. Additionally, this section establishes requirements for snubbers not able to perform its support function.

LCO 3.16.1 establishes actions to take if one or more snubber becomes inoperable. This specification satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). None of the described requirements of LCO 3.16.1 are applicable with the reactor vessel defueled, since all safety-related systems are no longer required to be operable with fuel permanently removed from the reactor vessel. As such, there are no safety-related snubbers necessary to mitigate the remaining DBA (i.e., FHA). LCO 3.16.1 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition. The snubber requirements no longer meet the criteria in 10 CFR 50.36(c)(2)(ii).

Summary:

The TS section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the shock suppressor (snubber) specification addressed in TS Section 3.16 will no longer apply or be required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.17 – REACTOR BUILDING AIR TEMPERATURE	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.17.1 – Primary Containment average air temperature	LCO 3.17.1 – Deleted
LCO 3.17.2 – Air temperature limits exceeded when critical	LCO 3.17.2 – Deleted
LCO 3.17.3 – Primary containment average air temperature calculation	LCO 3.17.3 – Deleted

BASIS

TS Section 3.17, Reactor Building Air Temperature, establishes specified temperature limits to ensure that the containment design temperature and pressure will not be exceeded in the event of a design basis loss of coolant accident. The limits also assure the maintenance of acceptable ambient environmental conditions for safety-related components located inside the containment. The containment air requirements satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO 3.17.1 establishes primary containment average air temperature limits. This temperature limit is applicable during power operations. LCO 3.17.1 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.17.2 establishes reactor shutdown requirements if the temperature limits of LCO 3.17.1 are exceeded. LCO 3.17.2 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.17.3 establishes how the primary containment average air temperature is to be calculated. This temperature limit is applicable during power operations. LCO 3.17.3 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

Summary:

The TS section is being proposed for deletion in its entirety. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and the reactor building air temperature specifications addressed in TS Section 3.17 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.19 – CONTAINMENT SYSTEMS	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.19.1 – Containment Structural Integrity LCO 3.19.2 – Previously Deleted	LCO 3.19.1 – Deleted LCO 3.19.2 – Deleted

BASIS

TS Section 3.19, Containment Integrity, establishes a requirement to verify containment structural integrity in accordance with the inservice tendon surveillance program for the reactor building prestressing system. These specifications satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). This LCO is associated with Surveillance Requirement (SR) 4.4.2, "Structural Integrity." The Inservice Tendon Surveillance Program for structural integrity and corrosion protection conforms to the recommendations of the NRC Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," and the requirements of Subsection IWL of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a. The detailed surveillance program for the prestressing system tendons is based on periodic inspection and mechanical tests to be performed on selected tendons.

LCO 3.19.1, Containment Structural Integrity, establishes controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. LCO 3.19.1 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

LCO 3.19.2 – Previously Deleted

Summary:

The TS section is being proposed for deletion in its entirety. This specification and SR does not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the containment structural integrity addressed in TS Section 3.19 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases and surveillance section will also be removed.

TS SECTION 3.20 – Previously Deleted

TS SECTION 3.21 RADIOACTIVE EFFLUENT INSTRUMENTATION – Previously Deleted

TS SECTION 3.22 RADIOACTIVE EFFLUENTS – Previously Deleted

TS SECTION 3.23 RADIOLOGICAL ENVIRONMENTAL MONITORING – Previously Deleted

These TS Sections were previously removed. The remaining references to these TS sections are proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 3.24 – REACTOR WATER LEVEL INDICATION	
Current TMI LCO	**Proposed TMI LCO**
LCO 3.24 – Reactor Vessel Water Level Indication	LCO 3.24 – Deleted

BASIS

TS Section 3.24, Reactor Water Level Indication, assures the operability of the Reactor Vessel Water Level Indication instrumentation that may be useful in diagnosing situations which could represent or lead to inadequate core cooling. This specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO 3.24 is applicable when the reactor is critical. LCO 3.24 is not proposed for inclusion in the PDTS since the TMI license will no longer be authorized for operation of the reactor in a permanently defueled condition.

Summary:

The section is being proposed for deletion in its entirety. This specification does not apply to the safe storage and handling of spent fuel in the SFP. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied and the Reactor Vessel Water Level Indication system addressed in TS Section 3.24 is no longer applicable or required. With the TS section deleted in its entirety the applicable base and surveillance section will also be removed.

TS SECTION 4 – SURVIELLANCE STANDARDS

TS Section 4 describes the SRs associated with the TS Section 3 LCOs. In accordance with 10 CFR 50.36(c)(3), surveillance requirements are related to testing, calibrating, or inspecting SSCs to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

Since there are no safety limits that apply to TMI with the reactor shutdown and defueled, and since there are relatively few remaining LCOs, the number of corresponding surveillance requirements has also been greatly reduced.

Due to the reduced number of LCOs and Surveillance Requirements, TMI proposes to combine the LCOs (TS Section 3) with the corresponding Surveillance Requirements (TS Section 4). This format will allow the Surveillance Requirements to be more readily associated with the corresponding LCO. The LCOs and combined SRs sections will be designated with notation 3/4.#. The proposed format to the LCO/SRs is shown in Attachment 2.

The list below contains a comparison between the provisions of the current TMI SR and the proposed PDTS. Each subsection of TMI TS Section 4 is discussed in more detail in the tables below.

Current TMI SR	**Proposed PDTS**
4.0 Surveillance Requirement Applicability	Applicable (Proposed New ***3/4.0 Limiting Conditions for Operations and Surveillance Requirement Applicability***)
4.1 Operational Safety Review	Deleted

4.2 Reactor Coolant System Inservice and Testing	Deleted
4.3 Deleted	Previously Deleted
4.4 Reactor Building	Deleted
4.5 Emergency Loading Sequence and Power Transfer, Emergency Core Cooling System & Reactor Building Cooling System Periodic Testing	Deleted
4.6 Emergency Power Periodic Testing	Deleted
4.7 Reactor Control Rod System Tests	Deleted
4.8 Deleted	Previously Deleted
4.9 Decay Heat Removal (DHR) Capability – Periodic Testing	Deleted
4.10 Reactivity Anomalies	Deleted
4.11 Reactor Coolant System Vents	Deleted
4.12 Air Treatment System	Deleted
4.13 Radioactive Materials Source Surveillance	Deleted
4.14 Deleted	Previously Deleted
4.15 Main Steam System Inservice Inspection	Deleted
4.16 Reactor Internals Vent Valves Surveillance	Deleted
4.17 Shock Suppressors (Snubbers)	Deleted
4.18 Deleted	Previously Deleted
4.19 Steam Generator (SG) Tube Integrity	Deleted
4.20 Reactor Building Air Temperature	Deleted
4.21 Radioactive Effluent Instrumentation	Previously Deleted
4.22 Radioactive Effluents	Previously Deleted
4.23 Radiological Environmental Monitoring	Previously Deleted

TS SECTION 4.0, SURVEILLANCE STANDARDS

TS Section 4.0 "Surveillance Standards," contain the general requirements applicable to all SRs and applies at all times unless otherwise stated in TSs. These specifications are referred to as "Surveillance Requirements (SR) Applicability" in in Standard Technical Specifications. They are requirements relating to testing, calibration, or inspection of SSCs to assure that the necessary quality of systems and components is maintained.

SRs 4.0.1 and SR 4.0.2 have been revised to reflect the permanently shutdown and defueled condition in the proposed PDTS. A new SR (4.0.3) is being proposed in the PDTS (see discussion in proposed TS Section 3/4.0). This SR is based on NUREG-1430, *"Standard Technical Specifications Babcock and Wilcox Plants"* (Reference 6) and Draft NUREG-1625, *"Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants"* (Reference 7), which has been modified to reflect the permanently defueled condition.

Current TMI SR	**Proposed TMI TS**
4.0.1 - During Reactor Operational Conditions for which a Limiting Condition for Operation (LCO) does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/ component operable, the associated surveillance requirement must be current. Failure to perform a surveillance within the specified Frequency shall be failure to meet the LCO except as provided in 4.0.2.	4.0.1 – Surveillance requirements shall be met during the specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in 4.0.2.

BASIS

SR 4.0.1 establishes the requirement that SRs must be met or current during Reactor Operational Conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification is to ensure that Surveillances are performed to verify the operability of systems and components, and that variables are within specified limits.

SR 4.0.1 is proposed for revision to remove references to reactor operational conditions. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the reference to operation conditions is no longer relevant and is therefore being deleted. SR 4.0.1 is being textually aligned with SR 3.0.1 in NUREG-1430 (Reference 6) and Draft NUREG-1625 (Reference 7), except that it is modified for a facility in permanently defueled condition.

TS 4.0.1 is relocated to proposed PDTS Section 3/4.0 as SR 4.0.1. This revision is editorial in nature.

Current TMI SR	**Proposed TMI SR**
4.0.2 - If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance.	4.0.2 - If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the

The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed. If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered. When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.	Surveillance. The delay period is only applicable when there is a reasonable expectation the surveillance will be met when performed. ~~A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.~~ If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered. When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.
BASIS	
SR 4.0.2 is textually aligned with SR 3.0.3 in Draft NUREG-1625 (Reference 7). The discussion of performing a risk evaluation for any ST delayed greater than 24 hours is eliminated. The diminished risk associated with a permanently shutdown and defueled facility do not necessitate the need for a detailed risk assessment for a missed SR. SR 4.0.2 is relocated to proposed PDTS Section 3/4.0 as SR 4.0.2. This revision is editorial in nature.	

TS SECTION 4.1 – OPERATIONAL SAFETY REVIEW	
Current TMI SR	**Proposed TMI SR**
SR 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.	SR 4.1.1 – Deleted
SR 4.1.2 – Equipment and sampling test shall be performed as detailed in Tables 4.1-2, 4.1-3, and 4.1-5.	SR 4.1.2 – Deleted
SR 4.1.3 – Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration <…> noted in Table 4.1-4.	SR 4.1.3 – Deleted
SR 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 <…>.	SR 4.1.4 – Deleted
BASIS	
SR 4.1, Operational Safety Review, establishes the minimum frequency and type of surveillances to be applied to unit equipment and items directly related to safety limits and LCOs. This section contains several system SRs that will no longer be required with the removal of TS LCOs 3.1 and 3.5 and system operability	

requirements. As discussed in the "Design Basis Accident Analysis Applicable to Proposed Change" section of this attachment, the postulated DBAs and events associated with reactor or power operation analyzed in UFSAR Chapter 14 are no longer applicable in the permanently defueled condition. The remaining DBAs do not require any of the listed SSCs to mitigate the consequences of an event.

SR 4.1.1 specifies in Table 4.1-1, Instrument Surveillance Requirements, 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. All of the surveillance test requirements from SR 4.1 and Table 4.1-1 are proposed to be deleted in their entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

Table 4.1-1 Instrument Surveillance Requirements:

- Items 1-13 relate to Reactor Protection System described in TS 2.3 and LCO 3.5.1 (Table 3.5-1, Function A). The TS and LCO related to RPS are proposed for deletion and as such the SRs will no longer be required.
- Items 45 and 46 relate to Reactor Protection Anticipated Reactor Trip Setpoints described in LCO 3.5.1 (Table 3.5-1, Function B). The LCO related to RPS are proposed for deletion and as such the SRs will no longer be required.
- Items 14-21 and items 43 and 44 relate to Engineered Safeguards Actuation System (ESAS) described in LCO 3.5.3 and LCO 3.5.1 (Table 3.5-1, Function C). These LCOs related to ESAS are proposed for deletion and as such the SRs will no longer be required.
- Items 25, 27, 29 and 30 relate to TS 3.3, Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems. TS 3.3 is proposed for deletion and as such the SRs will no longer be required.
- Item 51 relates to the Heat Sink Protection system that defines the operability requirements for the Emergency Feedwater System that are described in LCO 3.4.1 and LCO 3.5.1 (Table 3.5-1, Function D). The LCOs related to EFW are proposed for deletion and as such the SR will no longer be required.
- Items 23 and 24 relate to Control Rod Position Indication described in LCO 3.5.2, Control Rod Group and Power Distribution Limits. LCO 3.5.2 is proposed for deletion and as such the SRs will no longer be required.
- Item 34 relates to the Full Incore System used to measure axial power imbalance and QPT described in LCO 3.5.2, Control Rod Group and Power Distribution Limits. LCO 3.5.2 is proposed for deletion and as such the SR will no longer be required.
- Items 26, 47, 49, 50 and 52 relate to the Accident Monitoring instruments in LCO 3.5.5, Accident Monitoring Instrumentation, Table 3.5-2. LCO 3.5.5 is proposed for deletion and as such he SRs will no longer be required.
- Items 28 d-f and 37 relate to instrumentation used for measuring Reactor Coolant System Leakage required by LCO 3.1.6, Leakage. LCO 3.1.6 is proposed for deletion and as such the SRs will no longer be required.
- Item 37 relates to Containment Temperature required by LCO 3.17, Reactor Building Air Temperature. LCO 3.17 is proposed for deletion and as such the SR will no longer be required.
- Item 54 relates to Reactor Vessel Water Level required by LCO 3.24, Reactor Vessel Water Level Indication. LCO 3.24 is proposed for deletion and as such the SR will no longer be required.
- Item 28 a-c, 31, 32, 36, 40, 41, and 53 were previously deleted and no required further evaluation.

SR 4.1.2 specifies surveillance requirements for equipment and sampling test detailed in Tables 4.1-2, 4.1-3, and 4.1-5.

Table 4.1-2 Minimum Equipment Test Frequency:

- Items 1 and 2 relate to Control Rod testing as required by LCO 3.5.2. LCO 3.5.2 is proposed for deletion and as such the SRs will no longer be required.
- Item 3 relates to operability of the Pressurizer Safety Valves required by LCO 3.1.1.3. LCO 3.1.1 is proposed for deletion and as such the SRs will no longer be required.
- Item 4 relates to operability of the Main Steam Safety Valves required by LCO 3.4.1.2. LCO 3.4 is proposed for deletion and as such the SRs will no longer be required.
- Item 5 relates to performing Fuel Handling Interlock checks prior to the start of each refueling period. This SR is not tied directly to any current safety limit or LCO. The fuel handling interlocks are not credited to prevent or mitigate the impacts of a FHA, and do not meet any 10 CFR 36 criteria for inclusion into TS. A review of NUREG-1430, Standard Technical Specification Babcock and Wilcox Plants (Reference 6) found that these interlock checks were not included. This SR is proposed for deletion.
- Items 7 and 12 relate to measuring RCS Leakage and Primary to Secondary Leakage as required by LCO 3.1.6. LCO 3.1.6 is proposed for deletion and as such the SRs will no longer be required.
- Item 9 relates to a performing a functional test of the Spent Fuel Cooling System. This SR is not tied directly to a current safety limit or LCO. Per UFSAR Section 9.4, the SFP cooling system is designed to cool a full core offload when a total of 1,720 spent fuel assemblies (based on Cycle 24) are already residing in the SFPs. This specification will be performed prior to the offload for the permanent shutdown defueling (end of Cycle 22), and the functionality of the SFP cooling system will have been demonstrated. Following defueling, there will be no further additions of irradiated fuel assemblies into the pool and the residual heat in the pool will continue to decay over time. Therefore, the SR is proposed for deletion in the PDTS.
- Item 10 relates to the Intake Pump House Floor (IPSH) Silt Accumulation. This SR is not tied directly to a current safety limit or LCO. The IPSH silt accumulation relates to maintaining operability of the IPSH pumps in order to provide cooling for ECCS systems as described in TS LCO 3.3. Based on the remaining DBAs after permanent shutdown and defueling, there are no active safety system required to provide mitigation. Therefore, the SR is proposed for deletion in the PDTS.
- Item 11 relates to the operability of the Pressurizer Block Valve (RC-V-2) as required by LCO 3.1.12. LCO 3.1.12 is proposed for deletion and as such the SR will no longer be required.
- Items 6 and 8 were previous deleted and are not proposed for inclusion into the PDTS. This is an editorial change.

Table 4.1-3 Minimum Sampling Frequency:

- Item 1 relates to RCS chemistry as required by LCO 3.1.4 for RCS Activity and LCO 3.1.6 RCS Chemistry. LCOs 3.1.4 and 3.1.5 are proposed for deletion and as such the SRs will no longer be required.
- Items 2 and 3 relate to boron concentration of the Borated Water Storage Tank and Core Flooding Tank as required by LCO 3.3. LCO 3.3 is proposed for deletion and as such the Surveillances will no longer be required.

- Item 4 relates to the Spent Fuel Pool boron concentration as required by TS 5.4. The requirement for SFP boron concentration is proposed to be relocated to SR 4.1.2 supporting proposed TS 3.1.2 SFP Boron Concentration.
- Item 5 relates to Secondary Coolant Activity as required by LCO 3.13. LCO 3.13 is proposed for deletion and as such the SRs will no longer be required.
- Items 6 thru 12 were previous deleted and are not proposed for inclusion into the PDTS. This is an editorial change.

Table 4.1-4 Post Accident Monitoring Instrumentation:

- Items 1 and 2 relate to operability of the Core Flood Tanks and Reactor Building Emergency Sump pH Control System as required by LCO 3.3. LCO 3.3 is proposed for deletion and as such the SRs will no longer be required.

SR 4.1.3 specifies the surveillance requirement in Table 4.1-4 for the post-accident monitoring instrumentation as required by LCO 3.5.5 and Table 3.5-3. LCO 3.5.5 is proposed for deletion and as such the Surveillances will no longer be required.

Summary:

SR 4.1 is proposed to be deleted in its entirety with the exception of the SR related to SFP boron concentration which is proposed to be relocated to SR 4.1.2.1. Deletion of these SRs is acceptable since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), and none of the instruments or support equipment associated with the SRs in this section are required to function in the permanently defueled conditions which removes any associated requirements for testing. With these TS SRs deleted in its entirety the applicable bases section will also be also removed.

TS SECTION 4.2 – REACTOR COOLANT SYSTEM INSERVICE AND TESTING	
Current TMI SR	**Proposed TMI SR**
SR 4.2.1 – ISI of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a, except where specific written relief has been granted by the NRC.	SR 4.2.1 – Deleted
SR 4.2.2 – Previously Deleted	SR 4.2.2 – Deleted
SR 4.2.3 – Previously Deleted	SR 4.2.3 – Deleted
SR 4.2.4 – The accessible portions of one reactor coolant pump motor flywheel assembly will be ultrasonically inspected within the first ISI period, two reactor coolant pump motor flywheel assemblies within the first two ISI periods and all four by the end of the 10-year inspection interval. <…>	SR 4.2.4 – Deleted
SR 4.2.5 – Previously Deleted	SR 4.2.5 – Deleted
SR 4.2.6 – Previously Deleted	SR 4.2.6 – Deleted
	SR 4.2.7 – Deleted

SR 4.2.7 – A surveillance program for the pressure isolation valves between the primary coolant system and the low pressure injection system shall be as follows: <…>	
BASIS	
SR 4.2, Reactor Coolant System Inservice and Testing, establishes the SRs for the RCS and components subject to ASME XI boiler and pressure vessel code. As stated in ASME Section XI; *"The rules of this section constitute requirements to maintain the nuclear power plant and to return the plant to service, following plant outages, in a safe and expeditious manner."* This section contains several SRs that will no longer be required with the removal of TS LCO 3.1.6, LCO 3.3.2 and LCO 3.3.3. As discussed in the "Design Basis Accident Analysis Applicable to Proposed Change" section of this attachment, the postulated DBAs and events analyzed in UFSAR Chapter 14 are no longer applicable in the permanently defueled condition section of this enclosure. The remaining DBAs do not require any of the listed SSCs to mitigate the consequences of the event. <u>Summary</u>: All of the surveillance test requirements from SR 4.2 are proposed to be deleted in their entirety. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the provisions of maintenance of ASME Section XI requirements no longer needs to be described in this TS section. With the TS section deleted in its entirety the applicable bases section will also be removed.	

TS SECTION 4.3 Previously Deleted

This TS Section was previously removed. The remaining reference to this TS section is proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 4.4 – REACTOR BUILDING	
Current TMI SR	**Proposed TMI SR**
SR 4.4.1 – Containment Leakage Tests SR 4.4.2 – Structural Integrity SR 4.4.3 – Previously Deleted SR 4.4.4 – Previously Deleted	SR 4.4.1 – Deleted SR 4.4.2 – Deleted SR 4.4.3 – Deleted SR 4.4.4 – Deleted
BASIS	
SR 4.4, Reactor Building, establishes the SRs to supports the operability of containment and personnel airlock leakage, and structural integrity described in TS LCO 3.6 and LCO 3.19. TS LCO 3.6 and LCO 3.19 are not proposed for inclusion in the PDTS; therefore, these SRs are no longer required. SR 4.4.1 is no longer required with the removal of TS LCO 3.6. Under these conditions the provisions of 10 CFR Part 50 Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, no longer apply, as stated in 10 CFR Part 50.54(o), Condition of Licenses, "*Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under*	

§§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in Appendix J to this part.”

SR 4.4.2 is associated with LCO 3.19, which is proposed to be removed since containment tendon inspections are no longer required as discussed above in LCO 3.19.

SR 4.4.3 and SR 4.4.4 – Previously Deleted

Summary:

All of the surveillance test requirements from SR 4.4 are proposed to be deleted in their entirety. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the provisions for maintaining the Reactor Building no longer needs to be described in this TS section. With the TS section deleted in its entirety the applicable bases section will also be removed.

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

Current TMI SR	Proposed TMI SR
SR 4.5.1 – Emergency Loading Sequence	SR 4.5.1 – Deleted
SR 4.5.2 – Emergency Core Cooling Systems	SR 4.5.2 – Deleted
SR 4.5.3 – Reactor Building Cooling and Isolation System	SR 4.5.3 – Deleted
SR 4.5.4 – Engineered Safeguards Feature (ESF) Systems Leakage	SR 4.5.4 – Deleted

BASIS

SR 4.5, Emergency Loading Sequence and Power Transfer, Emergency Core Cooling System & Reactor Building Cooling System Periodic Testing, establishes the SRs for ensuring the operability of systems necessary to protect the reactor core, RCS, and containment systems in the event of a postulated DBA. These SRs support operability of TS LCO 3.3 and LCO 3.7, which are not proposed for inclusion in the PDTS.

SR 4.5.1, Emergency Loading Sequence, establishes the requirements to verify the emergency loading sequence and automatic power transfer that controls the operation of the pumps associated with the emergency core cooling system and Reactor Building cooling system are operable. All of the surveillance test requirements from SR 4.5.1 are proposed to be deleted in their entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

SR 4.5.2, Emergency Core Cooling Systems, ensures the high and low pressure injection systems and components and core flooding tanks are operable. 10 CFR 50.46(a)(1)(i) states *"This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted."* All of the surveillance test requirements from SR 4.5.2 are proposed to be deleted in their entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

SR 4.5.3, Reactor Building Cooling and Isolation System, establishes the requirements to verify the Reactor Building Spray Pump and components are operable. All of the surveillance test requirements from SR 4.5.3 are proposed to be deleted in their entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

SR 4.5.4, Engineered Safeguards Feature (ESF) Systems Leakage, establishes the requirements to ensure the leakage a low leakage rate from the ESF systems in LCO 3.3 in order to prevent significant offsite exposures and dose consequences. All of the surveillance test requirements from SR 4.5.4 are proposed to be deleted in their entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition.

Summary:

All of the surveillance test requirements from SR 4.5 are proposed to be deleted in their entirety. 10 CFR 50.46(a)(1)(i) states *"This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted."* Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the ECCS requirements are proposed for deletion and the SRs are no longer required. With the TS section deleted in its entirety the applicable bases section will also be removed.

TS SECTION 4.6 – EMERGENCY POWER SYSTEM PERIODIC TESTS	
Current TMI SR	**Proposed TMI SR**
SR 4.6.1 – Diesel Generators SR 4.6.2 – Station Batteries SR 4.6.3 – Pressurizer Heaters	SR 4.6.1 – Deleted SR 4.6.2 – Deleted SR 4.6.3 – Deleted
BASIS	

SR 4.6, Emergency Power Periodic Testing, establishes periodic testing and SRs of the emergency power systems to support the operability of the equipment specified in TS LCO 3.7, Unit Electric Power System. Since this section exists solely to support the emergency electrical system test requirements, the elimination of the need for the electrical systems also obviates the need for their support systems in the associated TS sections. LCO 3.7 is proposed to be deleted since the emergency power systems are no longer required to support the permanently defueled condition. Therefore, the SRs associated with demonstrating operability of LCO 3.7 are no longer required.

SR 4.6.1, Diesel Generators, establishes SR to demonstrate that one diesel generator will provide power for operation of safeguards equipment. The SR also assures 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 AC station service power or upon the receipt of an Engineered Safeguards Actuation Signal. The need for providing ECCS protection as defined in LCO 3.3 and 3.7 is proposed for deletion in PDTS since it is no longer required in a permanently defueled condition.

SR 4.6.2, Station Batteries, establishes SR to provide an indication of a cell becoming unserviceable long before it fails. The Station DC power distribution system as defined by LCO 3.7 is proposed for deletion in the PDTS since it is no longer required to support the permanently defueled condition.

SR 4.6.3, Pressurizer Heaters, ensures that a minimum of 107 kw of pressurizer heaters and their associated controls are capable of being supplied electrical power from an emergency bus to provide assure that these heaters can be energized during a loss of offsite power condition to maintain natural circulation. LCO 3.1.3.4.2 is proposed for deletion since this function is not required in the permanently defueled condition.

Summary: All of the surveillance test requirements from SR 4.6 are proposed to be deleted in their entirety since the LCOs they support are proposed for deletion. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the provisions to maintain emergency diesel generators, station batteries, and pressurizer heaters are no longer needed. With the TS section deleted in its entirety the applicable bases section will also be removed.

TS SECTION 4.7 – REACTOR CONTROL ROD SYSTEM TESTS	
Current TMI SR	**Proposed TMI SR**
SR 4.7.1 – Control Rod Drive System Functional Tests SR 4.7.2 – Control Rod Program Verification (Group vs. Core Positions)	SR 4.7.1 – Deleted SR 4.7.2 – Previously Deleted
BASIS	
SR 4.7, Reactor Control Rod Drive System Tests, establishes the SR to assure control rod operability. This SR support the operability of control rod as specified in TS LCO 3.5.2. LCO 3.5.2 is proposed for deletion Therefore, the SRs associated with demonstrating operability of LCO 3.5.2 are no longer required. Summary: All of the surveillance test requirements from SR 4.7 are proposed to be deleted in their entirety. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the requirements to ensure control rod operability no longer apply. With the TS section deleted in its entirety the applicable bases section will also be removed.	

TS SECTION 4.8 Previously Deleted
This TS Section was previously removed. The remaining reference to this TS section is proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 4.9 – DECAY HEAT REMOVAL (DHR) CAPABILITY – PERIODIC TESTING	
Current TMI SR	**Proposed TMI SR**
SR 4.9.1 – Reactor Coolant System (RCS) Temperature Greater than 250 degrees F	SR 4.9.1 – Deleted
SR 4.9.2 – RCS temperature less than or equal to 250 degrees F	SR 4.9.2 – Deleted

BASIS
SR 4.9, Decay Heat Removal (DHR) Capability – Periodic Testing, establishes the SR to support the operability of DHR as specified in TS LCO 3.4. LCO 3.4 is proposed for deletion Therefore, the SRs associated with demonstrating operability of LCO 3.4 are no longer required. SR 4.9.1, Reactor Coolant System (RCS) Temperature greater than 250 degrees F, ensures the operability of the EFW and Condensate Storage Tank systems and components. SR 4.9.1 is proposed to be deleted in its entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition. SR 4.9.2, RCS temperature less than or equal to 250 degrees F, ensures the operability of the DHR system and components. SR 4.9.2 is proposed to be deleted in their entirety since the TMI license will no longer be authorized for power operation in a permanently defueled condition. Summary: All of the surveillance test requirements from SR 4.9 are proposed to be deleted in their entirety. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The DHR system is not required the safe storage and handling of spent fuel in the SFP. The DHR cooling specifications addressed in TS Section 4.9 are no longer applicable or required. With the TS section deleted in its entirety the applicable bases will also be removed.

TS SECTION 4.10 – REACTIVITY ANOMALIES	
Current TMI SR	**Proposed TMI SR**
SR 4.10.1 – Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. <…>	SR 4.10.1 – Deleted
BASIS	
SR 4.10, Reactivity Anomalies, requires the evaluation of reactivity anomalies of a specified magnitude occurring during operation of the unit. This SR does not have an associated LCO. This SR is based on a vendor recommendation that validates reactor core parameters against predicted values. This SR is not proposed for inclusion in the PDTS since the reactor will be permanently defueled and operations will be prohibited. Summary: SR 4.10 is proposed to be deleted. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the evaluation of reactivity anomalies is no longer needed. With the TS section deleted in its entirety the applicable bases section will also be removed.	

TS SECTION 4.11 – REACTOR COOLANT SYSTEM VENTS	
Current TMI SR	**Proposed TMI SR**
SR 4.11.1 – Each reactor coolant system vent path shall be demonstrated OPERABLE once per refueling interval <…>.	SR 4.11.1 – Deleted
BASIS	
SR 4.11, Reactor Coolant System Vents, ensures the RCS vents are capable of venting non-condensable gases from the RCS. This SR support the operability of RCS vents as specified in TS LCO 3.1.13. LCO 3.1.13 is proposed for deletion Therefore, the SR associated with demonstrating operability of LCO 3.1.13 is no longer required. The basis is to ensure a method and system is available to remove steam and/or non-condensable gases from the RCS, which may inhibit core cooling during natural circulation. <u>Summary:</u> SR 4.11 is proposed to be deleted. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the need to establish core cooling by natural circulation is no longer needed. With the TS section deleted in its entirety the applicable bases section will also be removed.	

TS SECTION 4.12 – AIR TREATMENT SYSTEM	
Current TMI SR	**Proposed TMI SR**
SR 4.12.1 – Emergency Control Room Air Treatment System SR 4.12.2 – Previously Deleted SR 4.12.3 – Previously Deleted SR 4.12.4 – Fuel Handling Building ESF Air Treatment System	SR 4.12.1 – Deleted SR 4.12.2 – Deleted SR 4.12.3 – Deleted SR 4.12.4 – Deleted
BASIS	
SR 4.12, Air Treatment Systems, supports the operability of the equipment specified in TS LCO 3.15.1 for the emergency control room air treatment system and associated components and TS LCO 3.15.4 for the Auxiliary and Fuel Handling Building Air Treatment System and associated components. These LCOs are is proposed for deletion Therefore, the SRs associated with demonstrating operability of LCO 3.15.1 and LCO 3.15.4 are no longer required. SR 4.12.1 – Emergency Control Room Air Treatment System, establishes that this system and associated components will be able to perform its design function. SR 4.12.1 supports LCO 3.15.1 which is proposed for deletion and the SR required to support operability and is no longer required. SR 4.12.2 and SR 4.12.3 – Previously Deleted SR 4.12.4, Fuel Handling Building ESF Air Treatment System, establishes that this system and associated components will be able to perform its design function. SR 4.12.4 supports LCO 3.15.4 which is proposed for deletion and the SR required to support operability and is no longer required.	

Summary:
All of the surveillance test requirements from SR 4.12 are proposed to be deleted in their entirety. Since TS LCO 3.15 is proposed to be deleted, the SR required to support operability is no longer required. With the TS section deleted in its entirety the applicable bases section will also be removed.

TS SECTION 4.13 – RADIOACTIVE MATERIALS SOURCE SURVEILLANCE	
Current TMI SR	**Proposed TMI SR**
SR 4.13, Radioactive Materials Source Surveillance	SR 4.13 – Deleted
BASIS	
SR 4.13, Radioactive Materials Source Surveillance, supports the operability of TS LCO 3.10. LCO 3.10 is proposed for deletion; therefore, the SR is no longer required. Summary: SR 4.13 is not proposed for inclusion in the PDTS. This requirement is not credited in any safety analysis and does not meet any of the criteria in 10 CFR 50.36(c)(2)(ii). Further, this TS Section is not included in the Standardized TS provided in NUREG-1430 (Reference 6) or Draft NUREG-1625 (Reference 7). Therefore, this specification is proposed for deletion from the PDTS. Licensed sealed sources are controlled under a licensee controlled program.	

TS SECTION 4.14 – Previously Deleted
This TS Section was previously removed. The remaining reference to this TS section is proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 4.15 – MAIN STEAM SYSTEM INSERVICE INSPECTION	
Current TMI SR	**Proposed TMI SR**
SR 4.15.1 – The four weld joints <...> shall be 100 percent inspected in accordance with the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant components, defined in the TMI Inservice Inspection Program.	SR 4.15.1 – Deleted
BASIS	
SR 4.15, Main Steam System Inservice Inspection, establishes the SR to perform the inservice inspection of four welds in the Main Steam System identified as MS-0001, MS-0002, MS-0003, and MS-0004L of the TMI Inservice Inspection Program. The Inservice Inspection Program is to provide assurance of the continuing integrity of that portion of the Main Steam System in which a postulated failure would produce pressures in excess of the compartment wall and/or slab capacities. As discussed in the "Design Basis Accident Analysis Applicable to Proposed Change" section of this attachment, the postulated DBAs and events associated with reactor or power operation analyzed in UFSAR Chapter 14 are no longer applicable	

in the permanently defueled condition. As such, Main Steam System integrity is no longer applicable, which removes all requirements for testing.

Summary:

The surveillance test requirements from SR 4.15 are proposed to be deleted in their entirety. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the provisions of the TMI Inservice Inspection Program are no longer needed or required. With the TS section deleted in its entirety the applicable bases section will also be removed.

TS SECTION 4.16 – REACTOR INTERNALS VENT VALVES SURVEILLANCE	
Current TMI SR	**Proposed TMI SR**
SR 4.16.1 – Reactor Internals Vent Valves	SR 4.16.1 – Deleted

BASIS

SR 4.16, Reactor Internals Vent Valve Surveillance, supports the operability of TS LCO 3.1.11. This SR verifies vent valve freedom of movement to ensures 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. LCO 3.1.11 is proposed for deletion; therefore, the SR is no longer required.

Summary:

The surveillance test requirements from SR 4.16 are proposed to be deleted in their entirety. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel, pursuant to 10 CFR Part 50.82(a)(2). Therefore, the SR requirement to verify vent valve freedom of movement is no longer applicable or required. With the TS section deleted in its entirety the applicable bases section will also be removed.

TS SECTION 4.17 – SHOCK SUPPRESSORS (SNUBBERS)	
Current TMI SR	**Proposed TMI SR**
SR 4.17.1 – Each snubber shall be demonstrated OPERABLE by performance of the following inspection program.	SR 4.17.1 – Deleted

BASIS

SR 4.17, Shock Suppressor (Snubber), establishes the SR to examine, inspect, and functionally test the snubbers to assure its operability. This SR is associated with LCO 3.16, which is not proposed for inclusion in the PDTS since all systems associated with snubbers are no longer required to be operable with fuel permanently removed from the reactor vessel.

Summary:

The surveillance test requirements from SR 4.17 are proposed to be deleted in their entirety. Once TMI dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR

50.82(a)(2). Therefore, the SR for shock suppressor (snubber) will no longer apply or be required. With the TS section deleted in its entirety the applicable bases section will also be removed.

<table>
<tr><th colspan="2">TS SECTION 4.19 – STEAM GENERATOR (SG) TUBE INTEGRITY</th></tr>
<tr><th>Current TMI SR</th><th>Proposed TMI SR</th></tr>
<tr><td>SR 4.19.1 – Verify SG tube integrity in accordance with the Steam Generator Program.

SR 4.19.2 – Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program prior to exceeding an average reactor coolant temperature of 200°F following an SG tube inspection.</td><td>SR 4.19.1 – Deleted

SR 4.19.2 – Deleted</td></tr>
<tr><th colspan="2">BASIS</th></tr>
<tr><td colspan="2">SR 4.19, Steam Generator (SG) Tube Integrity, establishes SRs to assure the RCS boundary integrity of the SG tubes. The SRs are associated with LCO 3.1.1.2 and TS 6.19, Steam Generator Program. LCO 3.1.1.2 and TS 6.19 are not proposed for inclusion in the PDTS. As discussed in the "Design Basis Accident Analysis Applicable to Proposed Change" section of this attachment, the postulated DBAs and events associated with reactor or power operation analyzed in UFSAR Chapter 14 are no longer applicable in the permanently defueled condition. As such, SG tube integrity is no longer applicable, which removes all requirements for testing.

<u>Summary</u>:

The surveillance test requirements from SR 4.19 are proposed to be deleted in their entirety. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the requirements to maintain SG tube integrity no longer apply. With the TS section deleted in its entirety the applicable bases section will also be removed.</td></tr>
</table>

<table>
<tr><th colspan="2">TS SECTION 4.20 – REACTOR BUILDING AIR TEMPERATURE</th></tr>
<tr><th>Current TMI SR</th><th>Proposed TMI SR</th></tr>
<tr><td>SR 4.20.1 – When the reactor is critical, the reactor building temperature will <…></td><td>SR 4.20.1 – Deleted</td></tr>
<tr><th colspan="2">BASIS</th></tr>
<tr><td colspan="2">SR 4.20, Reactor Building Air Temperature, establishes the SR to support the operability of the reactor building air temperature in TS LCO 3.17. LCO 3.17 is not proposed for inclusion in the PDTS since the requirements are applicable when the reactor is critical.

<u>Summary</u>:

The surveillance test requirement from SR 4.20 is proposed to be deleted in its entirety. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into</td></tr>
</table>

the reactor vessel as provided in 10 CFR Part 50.82(a)(2), maintaining the reactor building air temperature is no longer necessary.

SR 4.21 Radioactive Effluent Instrumentation – Previously Deleted **SR 4.22 Radioactive Effluents** – Previously Deleted **SR 4.23 Radiological Environmental Monitoring** – Previously Deleted
These TS Sections were previously removed. The remaining reference to these TS sections are proposed to be removed as part of the editorial cleanup of the PDTS.

TS SECTION 5 – DESIGN FEATURES

The existing TS Section 5 "Design Features," provides information and design requirement associated with plant systems. Sections will be deleted or revised as described in each change basis.

TS 5.1, "Site," and TS 5.4, "New and Spent Fuel Storage Facilities" (renumbered to TS 5.2 and retitled to "~~New and~~ Spent Fuel Storage Facilities") will remain applicable with the reactor permanently defueled. TS 5.2, "Containment," and TS 5.3, "Reactor," are not proposed for inclusion in the PDTS since the TMI Part 50 license will no longer authorize placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The design features that do not apply in a defueled condition are being proposed for deletion.

Current TMI TS	Proposed TMI TS
TS 5.1 Site 5.1.1 The Three Mile Island Nuclear Station Unit 1 is located in an area of low population density about ten miles southeast of Harrisburg, PA. It is in Londonderry Township of Dauphin County, Pennsylvania, about two and one-half miles north of the southern tip of Dauphin County, where Dauphin is coterminal with York and Lancaster Counties. The station is located on an island approximately three miles in length situated in the Susquehanna River upstream from York Haven Dam. Figure 5-1 is an extended plot plan of the site showing the plant orientation and immediate surroundings. The Exclusion Area as defined in 10 CFR 100.3, is a 2,000 ft. radius, including portions of Three Mile Island, the river surface around it, and a portion of Shelley Island, which is owned by Exelon Generation Company, LLC. The minimum distance of 2,000 ft. occurs on the shore of the mainland in a due easterly direction from the plant as shown on Figure 5-1 for the Exclusion Area. Figure 5-3 showing the physical location of the fence defines the "Restricted Area" surrounding the plant. The minimum distance of the "Restricted Area" is	TS 5.1 Site 5.1.1 The Three Mile Island Nuclear Station Unit 1 is located in an area of low population density about ten miles southeast of Harrisburg, PA. It is in Londonderry Township of Dauphin County, Pennsylvania, about two and one-half miles north of the southern tip of Dauphin County, where Dauphin is coterminal with York and Lancaster Counties. The station is located on an island approximately three miles in length situated in the Susquehanna River upstream from York Haven Dam. Figure 5-1 is an extended plot plan of the site showing the plant orientation and immediate surroundings. The ***description of the*** Exclusion Area as defined in 10 CFR 100.3, is ***located in the Final Safety Analysis Report, as updated.*** ~~*a 2,000 ft. radius, including portions of Three Mile Island, the river surface around it, and a portion of Shelley Island, which is owned by Exelon Generation Company, LLC. The minimum distance of 2,000 ft. occurs on the shore of the mainland in a due easterly direction from the plant as shown on Figure 5-1 for the Exclusion Area. Figure 5-3 showing the physical location of the fence defines the "Restricted Area" surrounding the*~~

approximately 560 feet and is from the centerline of the TMI Unit 2 Reactor Building to a point on the westerly shoreline of Three Mile Island. The minimum distance to the outer boundary of the low population zone is two miles as shown on T.S. Figure 5-2, which also depicts the site topography for a radius of five miles. T.S. Figure 5-3 depicts the locations of gaseous effluent release points and liquid effluent outfalls (as tabularized on page 5-10), and the meteorological tower location (designated as 'weather tower' on the figure).	*~~plant. The minimum distance of the "Restricted Area" is approximately 560 feet and is from the centerline of the TMI Unit 2 Reactor Building to a point on the westerly shoreline of Three Mile Island. The minimum distance to the outer boundary of the low population zone is two miles as shown on T.S. Figure 5-2, which also depicts the site topography for a radius of five miles. T.S. Figure 5-3 depicts the locations of gaseous effluent release points and liquid effluent outfalls (as tabularized on page 5-10), and the meteorological tower location (designated as 'weather tower' on the figure).~~*
TS 5.2 Containment TS 5.2.1 Reactor Building TS 5.2.2 Reactor Building Isolation System	TS 5.2 – Deleted (TS 5.4 renumbered as TS 5.2) TS 5.2.1 – Deleted TS 5.2.2 – Deleted
TS 5.3 Reactor TS 5.3.1 Reactor Core TS 5.3.2 Reactor Coolant System	TS 5.3 – Deleted TS 5.3.1 – Deleted TS 5.3.2 – Deleted
TS 5.4 New and Spent Fuel Storage Facilities	TS 5.*~~4~~***2** ~~New and~~ Spent Fuel Storage Facilities *(Discussed below)*
TS 5.5 Air Intake Tunnel Fire Protection Systems	TS 5.5 – Previously Deleted

Basis

TS Section 5.1.1, Site, provides a description of the station site and location. An administrative change to remove excessive detail associated with the site boundary is being proposed. This information is located within the station's UFSAR. Figure 5-3 and the associated Table will be relocated to the UFSAR. This change is administrative and will provide a more consistent branch reference and does not change the technical content. Therefore, the proposed change to this section is acceptable.

TS Section 5.2, Containment, provides references to principal design parameters and applicable design codes for the Reactor Building, and design standards for penetrations not serving accident-consequence-limiting system. Since the TMI license no longer allows use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), no portions of this specification is applicable. Therefore, the deletion of this section in its entirety is acceptable.

TS 5.3, Reactor Core, provides a description and requirements regarding the reactor core, fuel assemblies and control rod assemblies, and the RCS. Because the TMI Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2), this TS section will not apply in a defueled condition and is being proposed for deletion.

TS 5.5, Air Intake Tunnel Fire Protection Systems, was previously deleted. The specification is proposed to be removed for TS clean-up. This change is editorial.

Current TMI TS	**Proposed TMI TS**
TS 5.4 New and Spent Fuel Storage Facilities TS 5.4.1 New Fuel Storage a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools. For the new fuel storage vault, the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions. The spacing in the new fuel storage vault is sufficient to maintain Keff less than 0.95 based on storage of fuel assemblies in clean unborated water or less than 0.98 based on storage in an optimum hypothetical low density moderator (fog or foam) for fuel assemblies with a nominal enrichment of 5.0 weight percent U235. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition. For Spent Fuel Pool "A", the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 11.1 inches in both directions for the Region I racks and 9.2 inches in both directions for the Region II racks. The spacing in the Spent Fuel Pool "A" storage locations for both Region I and II is adequate to maintain Keff less than 0.95. Region I will store fuel with a maximum 5.0 percent initial enrichment. Region II will store new fuel with low enrichment. When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition. <…> b. Deleted. c. New fuel may also be stored in shipping containers.	TS 5.~~4~~**2** ~~NEW AND~~ SPENT FUEL STORAGE FACILITIES TS 5.~~4~~**2**.1 ~~New~~ ***Spent*** Fuel Storage ~~a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools.~~ ~~For the new fuel storage vault, the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions. The spacing in the new fuel storage vault is sufficient to maintain Keff less than 0.95 based on storage of fuel assemblies in clean unborated water or less than 0.98 based on storage in an optimum hypothetical low density moderator (fog or foam) for fuel assemblies with a nominal enrichment of 5.0 weight percent U235. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition.~~ For Spent Fuel Pool "A", the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 11.1 inches in both directions for the Region I racks and 9.2 inches in both directions for the Region II racks. The spacing in the Spent Fuel Pool "A" storage locations for both Region I and II is adequate to maintain Keff less than 0.95. Region I will store fuel with a maximum 5.0 percent initial enrichment. ~~Region II will store new fuel with low enrichment.~~ When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppm~~b~~ must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition. <…> ~~b. Deleted.~~ ~~c. New fuel may also be stored in shipping containers.~~

TS 5.4.2 Spent Fuel Storage	~~TS 5.4.2 Spent Fuel Storage~~
a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.	a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.
b. Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.	b. ~~*Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.*~~
c. Deleted.	~~*c. Deleted.*~~
d. The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:	~~*d.*~~ The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:
<…>	<…>
f. DELETED	~~*f. DELETED*~~
g. When spent fuel assemblies are stored in the Spent Fuel Pool "A", Region II storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-4.	~~*g. When spent fuel assemblies are stored in the Spent Fuel Pool "A", Region II storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-4.*~~
h. When spent fuel assemblies are stored in the Spent Fuel Pool "B", storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-5.	~~*h. When spent fuel assemblies are stored in the Spent Fuel Pool "B", storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-5.*~~

Basis

TS 5.4, New and Spent Fuel Storage Facilities, is proposed to be renumbered as TS 5.2 due to the deletion of TS 5.2, Containment, and 5.3, Reactor. This change is editorial. The title of the specification is being revised to "Spent Fuel Storage Facilities" since TMI will no longer receive or possess new fuel after it has permanently shutdown and defueled.

TS 5.4.1, New Fuel Storage, is proposed to be renumbered to TS 5.2.1. and retitled as "Spent Fuel Storage" since TMI will no longer receive or possess new fuel. This change is editorial.

TS 5.4.1.a is proposed to be renumbered to TS 5.2.1. The specification establishes requirements regarding the design, use, and maintenance of spent fuel storage racks in the SFP and new fuel storage vault. The text describing the new fuel storage vault and storage of new fuel in Region II are proposed to be deleted since TMI will no longer receive new fuel after it has permanently shutdown and defueled. The notation of "ppmb" is revised to "ppm" since boron concentration is the specified parameter. This change is administrative and will provide a more accurate description without changing the technical content.

TS 5.4.1.b was previously deleted. The step is proposed to be removed for TS clean-up. This change is editorial.

TS 5.4.1.c is proposed to be deleted and removed from PDTS. New fuel will no longer be received or stored in shipping containers.

TS 5.4.2, Spent Fuel Storage, is proposed to be included under proposed TS 5.2.1.a. The title "5.4.2 Spent Fuel Storage" will be deleted.

TS 5.4.2.a will remain unchanged.

TS 5.4.2.b is proposed to be deleted. Proposed TS LCO 3.1.2 Spent Fuel Pool Boron Concentration will specify the required boron concentrations in the SFP. TS 5.4.2.d is proposed to be editorially renumbered as TS 5.2.1.b.

TS 5.4.2.c was previously deleted. The step is proposed to be removed for TS clean-up. This change is editorial.

TS 5.4.2.f was previously deleted. The step is proposed to be removed for TS clean-up. This change is editorial.

TS 5.4.2.g is proposed to be relocated. The specification and Figure 5-4, Minimum Burnup Requirements for Fuel in Region II of the Pool "A" Storage Racks, will be relocated to proposed TS LCO 3/4.1.3, Spent Fuel Assembly Storage.

TS 5.4.2.h is proposed to be relocated. The specification and Figure 5-5, Minimum Burnup Requirements for Fuel in the Pool "B" Storage Racks, will be relocated to proposed TS LCO 3/4.1.3, Spent Fuel Assembly Storage.

TS SECTION 6 – ADMINISTRATIVE CONTROLS

The existing TS Section 6, "Administrative Controls," contains provisions relating to organization and management, procedures, recordkeeping, review and audit, programs, and reporting necessary to assure operation of the facility in a safe manner.

Exelon submitted an LAR dated November 10, 2017 (Reference 9) that proposed revisions and removed certain requirements from Section 6.0 that are not applicable to the facility in a permanently defueled condition. Specifically, the amendment will revise TS Section 6.1, "Responsibility"; TS Section 6.2, "Organization"; TS Section 6.3, "Facility Staff Qualifications"; TS Section 6.4, "Training"; TS Section 6.6, "Reportable Event Action"; TS Section 6.7, "Safety Limit Violation"; and TS Section 6.8, "Procedures and Programs" to reflect the staffing and training requirements for operating staff when the facility is permanently defueled and other minor administrative changes. That proposed license amendment, once approved, will be effective upon the submittal of the certifications required by 10 CFR 50.82(a)(1)(i) and (ii), and will be implemented immediately after TMI is permanently defueled. The proposed revisions submitted in reference 9 are not discussed below.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, several of the TS Section 6.0 Specifications are no longer applicable. Therefore, the administrative controls that do not apply in a defueled condition are being proposed for deletion.

Current TMI TS	Basis for Change/Deletion
TS 6.8.4.a Radiological Environment Monitoring Program	The requirement for a Land Use Census (TS 6.8.4.a.2) and participation in an Interlaboratory Comparison Program (TS 6.8.4.a.3) are not proposed for inclusion in the PDTS. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, TMI will no longer be authorized to operate or retain fuel in the reactor vessel. An analysis of the FHA in the SFP indicated that radiological doses at the EAB and LPZ are within allowable limits of 10 CFR 50.67 after a 60-day fuel decay period following permanent reactor shutdown. There will no longer be a need to modify the

	radiological monitoring program for use of areas at or beyond the site boundary.
2. Land Use Census	The Land Use Census is currently controlled by the ODCM with reporting made to the NRC in the Annual Radiological Environmental Operating Report (AREOR). The Land Use Census satisfies the requirements of 10 CFR 50 Appendix I Section IV.B.3. The Land Use Census will be retained in the ODCM.
3. Interlaboratory Comparison Program	This program is currently controlled in the ODCM which directs reports be made to the NRC in the AREOR. The Interlaboratory Comparison Program satisfies the requirements of 10 CFR 50 Appendix I Section IV.B.2. This program will be retained in the ODCM.
TS 6.8.4.b Radiological Environment Monitoring Program	The proposed changes to these specifications reformats previously defined terms from upper case to lowercase letters. As discussed in TS Sections 1.22, the term "MEMBERS OF THE PUBLIC," is not proposed for inclusion in the PDTS as defined terms in this license amendment request. The standard convention of indicating defined terms in all capital letters has been adopted in the PDTS. Therefore, since this term is no longer defined within the context of the PDTS, it is being reformatted to lowercase. This change is editorial in nature. There are no other proposed changes to these TSs.
TS 6.8.5 Reactor Building Leakage Rate Testing Program	TS 6.8.5 is not proposed for inclusion in the PDTS. The Reactor Building Leakage Rate Testing Program, establishes the implementation of the leakage rate testing of the Reactor Building as required by 10 CFR Part 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program includes provisions for preventive maintenance, periodic visual inspections and integrated leak testing. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the provisions of 10 CFR Part 50 Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, no longer apply as stated in 10 CFR Part 50.54(o), Condition of Licenses, *"Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in Appendix J to this part."* Based on the above, the program associated with this section is no longer applicable. Therefore, the proposed deletion of this section in its entirety is acceptable.
TS 6.9.1 Routine Reports A. DELETED B. Annual Reports C. DELETED	The proposed change to TS 6.9.1.A-C is to eliminate term "DELETED" and renumber item "B" as "A". Item B.2 will be renumbered A.1; items B.3, 4, 5 will be deleted. This change is editorial in nature.

TS 6.9.2 DELETED	TS Section 6.9.2 was previously deleted and serves as a placeholder. Placeholder specification are being proposed for deletion. This change is editorial in nature.
TS 6.9.3 Annual Radiological Environmental Operating Report	TS 6.9.3.1 is being revised to remove the wording associated with the "operation of the unit" and replace it with "facility." Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), this change will provide a more accurate description of the station's condition. With the proposed deletion of TS 6.9.2, TS 6.9.3 and TS 6.9.3.1 are being proposed to be renumbered as TS 6.9.2 and TS 6.9.2.1.
TS 6.9.4 Annual Radioactive Effluent Release Report	TS 6.9.4.1 is being revised to remove the wording associated with the "operation of the unit" and replace it with "facility." Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), this change will provide a more accurate description of the station's condition. With the proposed deletion of TS 6.9.2, TS 6.9.4 and TS 6.9.4.1 are being proposed to be renumbered as TS 6.9.3 and TS 6.9.3.1.
TS 6.9.5 Core Operating Limits Report	TS 6.9.5 is not proposed for inclusion in the PDTS. The Core Operating Limits Report (COLR), is generated prior to each reload cycle and contains cycle specific core operating limits and coefficients. Since the TMI license no longer authorizes the use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the information associated the COLR is no longer applicable and the proposed deletion of this section in its entirety is acceptable.
TS 6.9.6 Steam Generator Tube Inspection Report	TS 6.9.6 is not proposed for inclusion in the PDTS. Steam Generator Tube Inspection Report is a report submitted to the NRC within 180 days following the completion of the inspection performed in accordance with TS 6.19, Steam Generator (SG) Program. TS 6.19 is being proposed to be deleted from the PDTS. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the program associated with this section is no longer applicable. Therefore, the proposed deletion of this section in its entirety is acceptable.
TS 6.10 Record Retention	The requirements for Record Retention are proposed to be deleted from PDTS on the basis that they are adequately address by the Quality Assurance Program (10 CFR 50, Appendix B, Criterion XVII) and because provisions relating to record keeping do not assure safe operation of a facility in a permanently defueled condition. Facility operations in a defueled facility are performed in accordance with approved written procedures. Facility records document appropriate station activities. Retention of the records provides document retrievability for review of compliance with requirements and regulations. Post-

	compliance review of records does not assure operation of the facility in a safe manner as activities described in these documents have already been performed. Numerous other regulations such as 10 CFR 20, Subpart L, and 10 CFR 50.71 also require retention of certain records related to operation of the facility. Thus, Record Retention will be maintained by the Decommissioning Quality Assurance Program.
TS 6.11 Radiation Protection Program	TS 6.11 is not proposed for inclusion in the PDTS. This program requires procedures to be prepared for personnel radiation protection consistent with the requirements of 10 CFR 20. The program is developed to ensure nuclear plant personnel safety and has no impact on nuclear safety. Additionally, nuclear plant personnel are not 'members of the public.' Thus, the principal operative standard in Section 182a of the Atomic Energy Act: 'health and safety of the public' does not apply. The Radiation Protection Program administrative control is proposed to be deleted from the PDTS. The program is not necessary to assure operation of the facility in a safe manner and can be relocated from the TS to the UFSAR. The requirement to have procedures to implement Part 20 and the requirement for periodic review of these procedures is addressed under 10 CFR 20 Subpart B – Radiation Protection Programs.
TS 6.13 Process Control Plan	The specification for the Process Control Plan (PCP) is proposed for deletion. The PCP implements the requirements of 10 CFR Parts 20, 61 and 71, and 49 CFR 171-172. The actions of TS 6.13 are required by regulation and it is not necessary to restate the requirements in the PDTS. The PCP is conducted under standard procedures with revisions approved by facility processes and program changes are reported to NRC by the Annual Radioactive Effluent Report. Thus, TS 6.13 is not proposed for inclusion in the PDTS.
TS 6.19 Steam Generator (SG) Program	TS 6.19 is not proposed for inclusion in the PDTS. Steam Generator (SG) Program, ensures SG tube integrity is maintained. As discussed in the "Design Basis Accident Analysis Applicable to Proposed Change" section of this attachment, the postulated DBAs and event associated with reactor or power operation analyzed in UFSAR Chapter 14 are no longer applicable in the permanently defueled condition. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the program associated with this section is no longer applicable. Therefore, the proposed deletion of this section in its entirety is acceptable.
TS 6.20 Control Room Envelope Habitability Program	TS 6.20 is not proposed for inclusion in the PDTS. This administrative program ensures CRE habitability is maintained such that, with an operable HVAC system, the CRE occupants can safely implement actions to control the reactor and mitigate accidents from within the control room envelope. Equipment control from the control room is no longer required to mitigate any of the remaining postulated accidents or events. As discussed in the "Design Basis Accident Analysis Applicable to Proposed Change" section of this attachment, with the plant in a

	permanently defueled state, the postulated accidents associated with reactor or power operation analyzed in UFSAR Chapter 14 are no longer credible. The remaining credible DBAs do not credit or require the use of the control room for mitigation. The discussion supporting the proposed deletion of LCO 3.15.1, Emergency Control Room Air Treatment System, provides further information on the bases for the removal of the CRE requirements. Since the TMI license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel as provided in 10 CFR Part 50.82(a)(2), the program associated with this section is no longer applicable. Therefore, the proposed deletion of this section in its entirety is acceptable. As previously discussed, TS 3.15 is not proposed for inclusion in the PDTS.
TS 6.21 Surveillance Frequency Control Program	TS 6.21 is not proposed for inclusion in the PDTS. 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. The requirements regarding the Surveillance Frequency Control Program (SFCP) are not proposed for inclusion in the PDTS. The remaining TS LCOs proposed for this PDTS contains three surveillance requirements, which can easily be controlled and maintained since they are so few. Therefore, there is no further need to maintain this program and it can be eliminated. Thus, TS 6.21 is not proposed for inclusion in the PDTS.

The proposed changes are shown on the marked-up TMI TS pages included as Attachment 2.

3.0 REGULATORY EVALUATION

3.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Exelon has determined that the proposed changes do not require any exemptions or relief from regulatory requirements.

10 CFR 50.82 *"Termination of license."*

"(a) For power reactor licensees —

(1)(i) When a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of § 50.4(b)(8);

(ii) Once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of § 50.4(b)(9) and; <…>

(2) Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR Part 50 license no

longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel."

By letter dated June 20, 2017 (Reference 1), Exelon provided formal notification to the NRC pursuant to 10 CFR 50.82(a)(1)(i) of Exelon's contingent determination to permanently cease operations at TMI on or about September 30, 2019.

10 CFR 50.36 *"Technical specifications."*

In 10 CFR 50.36, *"Technical specifications,"* the NRC established its regulatory requirements related to the content of TS. In doing so, the NRC placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the NRC noted that applicants were expected to incorporate into their TS *"those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity."* (Statement of Consideration, *"Technical Specification for Facility Licenses; Safety Analysis Reports,"* 33 FR 18610 (December 17, 1968).

Pursuant to 10 CFR 50.36, TS are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports. However, the rule does not specify the particular requirements to be included in a plant's TS.

Section 50.36 of 10 CFR provides four criteria to define the scope of equipment and parameters to be included in the TS LCOs. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the RCS pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during design bases accidents or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those structures, systems, and components (SSCs) shown to be significant to public health and safety. These criteria, which were subsequently codified in changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953), also pertain to the TS requirements for safe storage of spent fuel. A general discussion of these considerations is provided below.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for *"installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary."* Since no fuel will be present in the reactor, or the reactor coolant system pressurized at the TMI facility following permanent defueling, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for *"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."* The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some DBAs which continue to apply to a facility authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a facility with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The applicable DBAs for TMI in the permanently defueled condition are discussed within this proposed amendment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for an *"SSC that is part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."* The intent of this criterion is to capture into the TS only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that apply to a plant authorized only to handle, store, and possess nuclear fuel, as discussed in more detail within this proposed amendment, the remaining credible DBA (Post Permanent Shutdown FHA) that could result in damage to a fission product barrier does not assume the functioning of any active SSCs to mitigate the consequences of the DBA. None of the remaining TS LCOs meet Criterion 3.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for an *"SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."* The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. All of the accident sequences that previously dominated risk at TMI will no longer be applicable once the reactor is in the permanently shut down and defueled condition.

10 CFR 50.36(c)(2)(iii) states that *"A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section."* Since TMI was originally licensed on April 19, 1974 (prior to August 18, 1995), the rule does not require that the particular considerations in paragraph (c)(2)(ii) be included in the TMI's TS.

10 CFR 50.36(c)(5) "*Administrative Controls*." *"Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."*

The particular administrative controls to be included in the TS generally are requirements the NRC deems necessary to support the safe operation of a facility that are not already covered by other regulations. Accordingly, the NRC staff determined that administrative control requirements that are not specifically required under Section 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or an event giving rise to an immediate threat to the public health and safety, may be relocated to more appropriate documents (e.g., Quality Assurance Program, Security Plan, or Emergency Plan), which are subject to regulatory controls. Similarly, while the required content of TS administrative controls is specified in 10 CFR Part 50.36(c)(5), particular details may be relocated to licensee-controlled documents, where other regulations provide adequate regulatory control.

10 CFR 50.36(c)(6) "*Decommissioning.*" *"This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications involving safety*

limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis."

As noted above, by letter dated June 20, 2017 (Reference 1), Exelon provided formal notification to the NRC pursuant to 10 CFR 50.82(a)(1)(i) of Exelon's determination to permanently cease operations at TMI on or about September 30, 2019. Upon submittal of the final certification that fuel has been permanently removed from the TMI reactor vessel pursuant to 10 CFR 50.82(a)(1)(ii), TMI will no longer be licensed to operate the reactor, or emplace or retain fuel in the reactor vessel. The proposed amendment deletes the portions of the previous TMI TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shutdown condition.

10 CFR 50.46 *"Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."*

"(a)(1)(i) <…> *This section does not apply to a nuclear power reactor facility for which the certifications required under 10 CFR 50.82(a)(1) have been submitted."*

10 CFR 50.48 *"Fire Protection."*

"(f) Licensees that have submitted the certifications required under § 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard). A fire protection program that complies with NFPA 805 shall be deemed to be acceptable for complying with the requirements of this paragraph.

(1) The objectives of the fire protection program are to--

(i) Reasonably prevent these fires from occurring;

(ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and

(iii) Ensure that the risk of fire-induced radiological hazards to the public, environment and plant personnel is minimized.

(2) The licensee shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities."

In 10 CFR 50.48(f), the NRC established the requirement for maintaining a fire protection program once a licensee has submitted the certifications required under 10 CFR 50.82(a)(1). Since the initial certification has been submitted pursuant to 10 CFR 50.82(a)(1)(i) (Reference 1) and once the final certification required by 10 CFR 50.82(a)(1)(ii) has been submitted, the requirements of 10 CFR 50.48(f) will be in full effect.

10 CFR 50.51 *"Continuation of license."*

"(b) Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the NRC notifies the licensee in writing that the license is terminated. During such period of continued effectiveness the licensee shall--

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and

(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."

Exelon will continue to conduct activities in accordance with the license until the NRC notifies Exelon in writing that the license is terminated.

10 CFR 50.62 *"Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."*

"(a) *Applicability. The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted."*

10 CFR 50.67 *"Accident source term."*

"(a) Applicability. The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997, and holders of renewed licenses under part 54 of this chapter whose initial operating license was issued prior to January 10, 1997, who seek to revise the current accident source term used in their design basis radiological analyses.

(b) Requirements. (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[1] previously analyzed in the safety analysis report.

(2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would

"[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

not receive a radiation dose in excess of 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

10 CFR 50.2 *"Definitions."* ***[Emphasis added]***

*"**Certified fuel handler** means, for a nuclear power reactor facility, a non-licensed operator who has qualified in accordance with a fuel handler training program approved by the Commission."*

*"**Responsible officer** means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or other individual in the organization of a corporation, partnership, or other entity who is vested with executive authority over activities subject to this part."*

*"**Safety-related structures, systems and components** means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:*

(1) The integrity of the reactor coolant pressure boundary

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable."

By letter dated December 29, 2017 (Reference 20), the NRC approved the Certified Fuel Handler training program for TMI.

3.2 Precedent

The proposed changes are consistent with the intent of the license and accompanying PDTS issued to the following facilities that have been permanently shutdown and defueled: (1) Crystal River Nuclear Plant, Unit 3, for which an amendment was issued on September 4, 2015 (Reference 5); (2) Fort Calhoun Station, Unit 1, for which an amendment was issued on March 6, 2018 (Reference 2); (3) Vermont Yankee Nuclear Power Station, for which an amendment was

"[2] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation."

issued on October 7, 2015 (Reference 3); (4) Kewaunee Power Station, for which an amendment was issued on February 13, 2015 (Reference 4).

3.3 No Significant Hazards Consideration (NSHC)

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon), proposes an amendment to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specifications (TS), of RFOL No. DPR-50 for Three Mile Island Nuclear Station (TMI).

The proposed amendment would revise the RFOL and the associated TS to a Renewed Facility License and Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The proposed amendment also revises the Current Licensing Basis (CLB) mitigation strategies for Flood Mitigation and Aircraft Impact Protection in the Air Intake Tunnel.

The proposed changes would revise and remove certain requirements contained within the RFOL and TS and remove the requirements that would no longer be applicable once it has been certified that all fuel has permanently been removed from the TMI reactor pursuant to 10 CFR 50.82(a)(1)(ii). Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are docketed, the 10 CFR Part 50 license for TMI no longer will authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). The proposed changes to the RFOL and TS not proposed for inclusion in the PDTS or revision are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages and sections, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS table of contents is also accordingly revised.

The existing TMI TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled plant.

Exelon has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes would not take effect until TMI has certified to the NRC that it has permanently ceased operation and entered a permanently defueled condition. Because the 10 CFR Part 50 license for TMI will no longer authorize operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR Part 50.82(a)(1) submitted, as specified in 10 CFR Part 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible.

The remaining UFSAR Chapter 14 postulated design basis accident (DBA) events that could potentially occur at a permanently defueled facility would be a Fuel Handling Accident (FHA) in the Spent Fuel pool (SFP), Waste Gas Tank Rupture (WGTR), and Fuel Cask Drop Accident (FCDA). The FHA analyses for TMI shows that, following 60 days of decay time after reactor shutdown and provided the SFP water level requirements of proposed TS LCO 3/4.1.1 are met, the dose consequences are acceptable without relying on SSCs to remain functional for accident mitigation during and following the event. The one exception to this is the continued function of the passive SFP structure. The remaining DBAs that support permanently shutdown and defueled condition do not rely on any active safety system for mitigation.

The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition and safe storage and handling of fuel will be the only operations performed, and therefore, bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation will no longer be credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes to delete and/or modify certain TMI RFOL, TS, or CLB have no impact on facility SSCs affecting the safe storage of spent irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of spent irradiated fuel itself. The removal of TS that are related only to the operation of the nuclear reactor, or only to the prevention, diagnosis, or mitigation of reactor related transients or accidents, cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shutdown and defueled and TMI will no longer be authorized to operate the reactor.

The proposed modification or deletion of requirements of the TMI RFOL, TS, and CLB do not affect systems credited in the accident analysis for the remaining credible DBAs at TMI. The proposed RFOL and PDTS will continue to require proper control and monitoring of safety significant parameters and activities. The TS regarding SFP water level and spent fuel storage is retained to preserve the current requirements for safe storage of irradiated fuel.

The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding, spent fuel racks, SFP integrity, and SFP water level). Since extended operation in a defueled condition and safe fuel handling will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes involve deleting and/or modifying certain RFOL, TS, and CLB once the TMI facility has been permanently shutdown and defueled. Because the 10 CFR Part 50 license for TMI no longer authorizes operation of the reactor, or emplacement or retention of fuel into the reactor vessel with the certifications required by 10 CFR Part 50.82(a)(1) submitted, as specified in 10 CFR Part 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. The remaining postulated DBA events that could potentially occur at a permanently defueled facility would be a FHA, WGTR, and FCDA. The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses.

The proposed changes are limited to those portions of the RFOL, TS, and CLB that are not related to the safe storage of irradiated fuel. The requirements that are proposed to be revised or deleted from the RFOL, TS, and CLB are not credited in the existing accident analysis for the remaining applicable postulated accidents; and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated design basis accidents involving the reactor will no longer be possible because the reactor will be permanently shutdown and defueled and TMI will no longer be authorized to operate the reactor.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above, Exelon concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

3.4 <u>Conclusion</u>

In conclusion, based on the considerations discussed above: 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 NRC'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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR Part 51.22(c)(9) as follows. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

In addition, the proposed changes involve changes to recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

5.0 REFERENCES

1. Letter from J. Bradley Fewell (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Certification of Permanent Cessation of Power Operations for Three Mile Island Nuclear Station, Unit 1," dated June 20, 2017 (ADAMS Accession No. ML17171A151)
2. Letter from U.S. Nuclear Regulatory Commission to Omaha Public Power District (OPPD), "Fort Calhoun Station, Unit 1 - Issuance of Amendment RE: Revised Technical Specifications to Align to Those Requirements for Decommissioning (CAC NO. MF9567; EPID L-2017-LLA-0192)," dated March 6, 2018 (ADAMS Accession No. ML18010A087)
3. Letter from U.S. Nuclear Regulatory Commission to Entergy Nuclear Operations, Inc., "Vermont Yankee Nuclear Power Station - Issuance of Amendment for Defueled Technical Specifications and Revised License Conditions for Permanently Defueled Condition (CAC No. MF3714)," dated October 7, 2015 (ADAMS Accession No. ML15117A551)
4. Letter from U.S. Nuclear Regulatory Commission to Dominion Energy Kewaunee, Inc., "Kewaunee Power Station - Issuance of Amendment for Permanently Shutdown and Defueled Technical Specifications and Certain License Conditions (TAC No. MF1952)," dated February 13, 2015 (ADAMS Accession No. ML14237A045)
5. Letter from U.S. Nuclear Regulatory Commission to Crystal River Nuclear Plant, "Crystal River Unit 3 Nuclear Generating Plant – Issuance of Amendment for Permanently Shutdown and Defueled Operating License and Technical Specifications (TAC No MF3089)," dated September 4, 2015 (ADAMS Accession No. ML15224B286)
6. NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants," Revision 4 (ADAMS Accession No. ML12100A177)
7. NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," Draft for Comment March 1998 (ADAMS Accession No. ML082330233)
8. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 - Approval of Certified Fuel Handler Training and Retraining Program (CAC NO. MF9960; EPID L-2017-LLL-0013)," dated December 29, 2017 (ADAMS Accession No. ML17228A729)
9. Letter from Michael P. Gallagher (Exelon Generation Company, LLC), to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Changes to Technical Specification Section 1.0, "Definitions," and 6.0, "'Administrative Controls" for Permanently Defueled Condition," dated November 10, 2017 (ADAMS Accession No. ML17314A024)
10. Calculation C-1101-900-E0000-088, "Fuel Handling Accident Dose Consequence (Post Permanent Shutdown)," Revision 0, dated May 11, 2018

11. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000, (ADAMS Accession No. ML003716792)

12. Letter from U.S. Nuclear Regulatory Commission to Christopher M. Crane (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment Regarding Relocation of Technical Specification Requirements for Refueling and Spent Fuel Pool Area Radiation Monitors (TAC NO. MD3783)," dated December 29, 2017 (ADAMS Accession No. ML072340348)

13. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Oyster Creek Nuclear Generating Station and Independent Spent Fuel Storage Installation - Review and Acceptance of Changes RE: Decommissioning Quality Assurance Program (EPID L-2017-LLQ-0003)," dated June 27, 2018 (ADAMS Accession No. ML18165A136)

14. Calculation C-1101-202-E410-476, "Spent Fuel Pool Thermohydraulic Analysis," Revision 0, Purpose 3, dated March 6, 2018

15. NRC Safety Evaluation by the Office of Nuclear Reactor Regulation, Direct Transfer of Facility Operating Licenses from Amergen Energy Company, LLC to Exelon Generation Company, LLC, License Nos. NPF-62, DPR-16 and DPR-50 Clinton Power Station, Unit No. 1; Oyster Creek Nuclear Generating Station; and Three Mile Island Nuclear Station, Unit 1; Docket Nos. 50-461, 50-219, 72-15 and 50-289, dated December 23, 2008 (ADAMS Accession No. ML082750072)

16. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U.S Nuclear Regulatory Commission, "Report on Status of Decommissioning Funding for Reactor," dated March 31, 2009 (ADAMS Accession No. ML090900463)

17. NUREG-1928, "Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1" issued October 2009 (ADAMS Accession No. ML092950450)

18. Letter from Pamela B. Cowan (Exelon Generation, LLC) to U.S. Nuclear Regulatory Commission – "Amended Submittal to Revision 20 of the Three Mile Island, Unit 1 Updated Final Safety Analysis Report," dated December 8, 2010 (ADAMS Accession No. ML110420396)

19. Letter from R. W. Libra (Exelon Generation, LLC) to U.S. Nuclear Regulatory Commission – "Notification of Implementation of License Renewal Activities in Preparation for Entrance into Period of Extended Operation for Unit 1," dated April 11. 2014 (ADAMS Accession No. ML14106A034)

20. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 - Approval of Certified Fuel Handler Training and Retraining Program (CAC NO. MF9960; EPID L-2017-LLL-0013)," dated December 29, 2017 (NRC Accession No. ML17228A729)

21. Letter from U.S. Nuclear Regulatory Commission to Bryan C. Hanson (Exelon Generation Company, LLC), "Three Mile Island Nuclear Station, Unit 1 - Staff Assessment of Response to 10 CFR 50.54(F) Information Request- Flood-Causing

Mechanism Re-Evaluation (CAC NO. MF1113)," dated November 2, 2017 (NRC Accession No. ML17276A218)

Attachment 2

Proposed Technical Specifications (Marked-Up Pages)

Three Mile Island Nuclear Station, Unit 1

Renewed Facility Operating License No. DPR-50

NRC Docket No. 50-289

Changes to the Renewed Facility Operating License (RFOL) and Appendix A, Technical Specification (TS)

(68 pages)

EXELON GENERATION COMPANY, LLC

(Three Mile Island Nuclear Station, Unit 1)

DOCKET NO. 50-289

RENEWED FACILITY ~~OPERATING~~ LICENSE

Renewed License No. DPR-50

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

   a. The application for a renewed license filed by the applicant 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 1 and all required notifications to other agencies or bodies have been duly made;

   b. DELETED~~Construction of the Three Mile Island Nuclear Station, Unit 1 (TMI or the facility) has been substantially completed in conformity with Construction Permit No: CPPR-40, the application, as amended, the provisions of the Act and the rules and regulations of the Commission;~~

   c. The facility will be maintained ~~operate~~ in conformity with the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   d. There is a reasonable assurance: (1) that the activities authorized by this renewed ~~operating~~ license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   e. Exelon Generation Company, LLC (Exelon Generation Company) is technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

   f. Exelon Generation Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   g. The issuance of this renewed ~~operating~~ license will not be inimical to the common defense and security or to the health and safety of the public;

h. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Renewed Facility ~~Operating~~ License No. DPR-50 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;

i. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including 10 CFR Section 30.33, 40.32, 70.23 and 70.31; and

j. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility=s current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

2. Renewed Facility ~~Operating~~ License No. DPR-50 is hereby issued to Exelon Generation Company to read as follows:

a. This renewed license applies to the Three Mile Island Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility), owned ~~and operated~~ by Exelon Generation Company. The facility is located in Dauphin County, Pennsylvania, and is described in the "Updated Final Safety Analysis Report (UFSAR)" as supplemented and amended and the Environmental Report as supplemented and amended.

b. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Exelon Generation Company, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess~~,~~ and use~~, and operate~~ the facility as required for fuel storage in accordance with the procedures and limitations set forth in this renewed license;

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

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess at either TMI-1 or TMI-2, and use in amounts as required for TMI-1 any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, testing, instrument calibration, or associated with radioactive apparatus or components. Other than radioactive apparatus and components to be used at TMI Unit 2 in accordance with the TMI-2 License, the radioactive apparatus and components that may be moved from TMI Unit 1 to TMI Unit 2 under this provision shall be limited to: (1) outage-related items (such as contaminated scaffolding, tools, protective clothing, portable shielding and decontamination equipment); and (2) other equipment belonging to TMI Unit 1 when storage of such equipment at TMI-2 is deemed necessary for load handling or contamination control considerations;

(4) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess at the TMI Unit 1 or Unit 2 site, but not separate, such byproduct and special nuclear materials ~~as may be~~ that were produced by the operation of either unit. Radioactive waste may be moved from TMI Unit 2 to TMI Unit 1 under this provision for collection, processing (including decontamination), packaging, and temporary storage prior to disposal. Radioactive waste that may be moved from TMI Unit 1 to TMI Unit 2 under this provision shall be limited to: (1) dry active waste (DAW) temporarily moved to TMI Unit 2 during waste collection activities, and (2) contaminated liquid contained in shared system piping and tanks. Radioactive waste that may be moved from TMI Unit 1 to TMI Unit 2 under this provision shall not include spent fuel, spent resins, filter sludge, evaporator bottoms, contaminated oil, or contaminated liquid filters.

The storage of radioactive materials or radwaste generated at TMI Unit 2 and stored at TMI Unit 1 shall not result in a source term that, if released, would exceed that previously analyzed in the UFSAR in terms of off-site dose consequences.

The storage of radioactive materials or radwaste generated at TMI Unit 1 and stored at TMI Unit 2 shall not result in a source term that, if released, would exceed that previously analyzed in the PDMS SAR for TMI Unit 2 in terms of off-site dose consequences.

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

(1) DELETED~~Maximum Power Level~~

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

(2) <u>Technical Specifications</u>

The Technical Specifications contained in ~~Appendices~~ Appendix A, as revised through Amendment No. ~~293~~[###], are hereby incorporated in the license. The Exelon Generation Company shall ~~operate~~ maintain the facility in accordance with the Permanently Defueled Technical Specifications (PDTS).

(3) <u>Physical Protection</u>

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.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 275 and modified by License Amendment No. 288

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

~~a. Identification of a sample 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.

~~e. Procedures defining corrective actions of off control point chemistry conditions; and~~

~~f. A procedure identifying (1) the authority responsible for the interpretation of the data, and (2) the sequence and timing of administrative events required to initiate corrective action.~~

(6) Inservice Testing - DELETED

(7) Aircraft Movements - DELETED

(8) Repaired Steam Generators - DELETED

(9) Long Range Planning Program - DELETED

Sale and License Transfer Conditions

(10) DELETED

(11) DELETED

(12) DELETED

(13) DELETED

(14) DELETED

(15) Exelon Generation Company shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application, the requirements of the Order Approving Transfer of License and Conforming Amendment, dated January 8, 2009, and the related Safety Evaluation dated December 23, 2008.

(16) DELETED

(17) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(18) DELETED~~Upon implementation of Amendment No. 264 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Specification 4.12.1.5, in accordance with TS 6.20.c.(i), the assessment of CRE habitability as required by Specification 6.20.c.(ii), and the measurement of CRE pressure as required by Specification 6.20.d, shall be considered met. Following implementation:~~

~~(a) The first performance of Specification 4.12.1.5, in accordance with Specification 6.20.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of Specification 1.25, as measured from August 21, 2000, the date of the most recent successful tracer gas test, as stated in the December 9, 2003, letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.~~

~~(b) The first performance of the periodic assessment of CRE habitability, Specification 6.20.c.(ii), shall be within 3 years, plus the 9-month allowance of Specification 1.25, as measured from August 21, 2000, the date of the most recent successful tracer gas test, as stated in the December 9, 2003, letter response to Generic Letter 2003-01 , or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.~~

~~(c) The first performance of the periodic measurement of CRE pressure, Specification 6.20.d, shall be within 24 months, plus the 180 days allowed by Specification 1.25, as measured from December 9, 2006, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.~~

(19) DELETED~~At the time of the closing of the transfer of TM1-1, and the respective license from AmerGen Energy Company, LLC (AmerGen) to Exelon Generation Company, AmerGen shall transfer to Exelon Generation Company ownership and control of AmerGen TMI NQF, LLC, and AmerGen Consolidation, LLC shall be merged into Exelon Generation Consolidation, LLC. Also at the time of the closing, decommissioning funding assurance provided by Exelon Generation Company, using an additional method allowed under 10 CFR 50.75 if necessary, must be equal to or greater than the minimum amount calculated on that date pursuant to, and required by 10 CFR 50.75 for TMI-1. Furthermore, funds dedicated for TMI-1 prior to closing shall remain dedicated to TMI-1 following the closing. The name of AmerGen TMI NQF, LLC shall be changed to Exelon Generation TMI NQF, LLC at the time of the closing.~~

(20) DELETED~~The information in the UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be incorporated into the UFSAR no later than the next scheduled update required by 10 CFR 50.71(e) following the issuance of this renewed operating license. Until this update is complete, Exelon Generation Company may not make changes to the information in the supplement. Following incorporation into the UFSAR, the need for prior Commission approval of any changes will be governed by 10 CFR 50.59.~~

(21) DELETED~~The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to and/or during the period of extended operation. The licensee shall complete these activities in accordance with Appendix A of NUREG-1928, "Safety Evaluation Report Related to the License Renewal of Three Mile Island, Unit 1," dated, October, 2009. The licensee shall notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection.~~

(22) Handling of irradiated fuel in the Spent Fuel Pool will not be permitted following implementation of the PDTS until a minimum of 60 days following the permanent shutdown.

d. This license is effective as of the date of issuance and ~~shall expire at midnight, April 19, 2034~~ is effective until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: October 22, 2009

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

~~LIST OF TABLES~~

LIST OF FIGURES

1. DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 ~~RATED POWER~~ACTIONS

~~Rated power is a steady state reactor core output of 2568 MWt.~~ACTIONS shall be that part of a Specification that prescribes required actions to be taken under designated Conditions within specified completion times.

1.2 ~~REACTOR OPERATING CONDITIONS~~ CERTIFIED FUEL HANDLER

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Specification 6.3.2.

1.3 NON-CERTIFIED OPERATOR

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 6.3.1, but is not a CERTIFIED FUEL HANDLER.

~~1.2.1 COLD SHUTDOWN~~

~~The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and Tave is no morethan 200°F. Pressure is defined by Specification 3.1.2.~~

~~1.2.2 HOT SHUTDOWN~~

~~The reactor is in the hot shutdown condition when it is subcritical by at least one percent delta k/k and Tave is ator greater than 525°F.~~

~~1.2.3 REACTOR CRITICAL~~

~~The reactor is critical when the neutron chain reaction is self sustaining and Keff = 1.0.~~

~~1.2.4 HOT STANDBY~~

~~The reactor is in the hot standby condition when all of the following conditions exist:~~

~~a. Tave is greater than 525°F~~

~~b. The reactor is critical~~

~~c. Indicated neutron power on the power range channels is less than two percent of rated power~~

~~1.2.5 POWER OPERATION~~

~~The reactor is in a power operating condition when the indicated neutron power is above two percent of rated power as indicated on the power range channels.~~

~~1.2.6 REFUELING SHUTDOWN~~

~~The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.~~

~~1.2.7 REFUELING OPERATION~~

~~An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.~~

~~1.2.8 REFUELING INTERVAL~~

~~Time between normal refuelings of the reactor. This is defined as once per 24 months.~~

~~1.2.9 STARTUP~~

~~The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.~~

~~1.2.10 Tave~~

~~Tave is defined as the arithmetic average of the coolant temperatures in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made.~~

~~1.2.11 HEATUP COOLDOWN MODE~~

~~The heatup-cooldown mode is the range of reactor coolant temperature greater than 200°F and less than 525°F.~~

~~1.2.12~~1.5 STATION, UNIT, PLANT, AND FACILITY

Station, unit, plant, and facility as used in these technical specifications all refer to TMI Unit 1.

1.4~~3~~ OPERABLE

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

~~1.4 PROTECTION INSTRUMENTATION LOGIC~~

~~1.4.1 INSTRUMENT CHANNEL~~

~~An instrument channel is the combination of sensor, wires, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control, and/or protection. An instrument channel may be either analog or digital.~~

~~1.4.2 REACTOR PROTECTION SYSTEM~~

~~The reactor protection system is described in Section 7.1 of the Updated FSAR. Itis that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.~~

~~1.4.3 PROTECTION CHANNEL~~

~~A PROTECTION CHANNEL as described in Section 7.1 of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.~~

~~1.4.4 REACTOR PROTECTION SYSTEM LOGIC~~

~~This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as described in Section 7.1 of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one out of two times two logic. Each element of the one out of two times two logic is controlled by a separate set of two out of four logic contacts from the four reactor protection channels.~~

~~1.4.5 ENGINEERED SAFETY FEATURES SYSTEM~~

~~This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.~~

~~1.4.6 DEGREE OF REDUNDANCY~~

~~The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.~~

~~1.5 INSTRUMENTATION SURVEILLANCE~~

~~1.5.1 TRIP TEST~~

~~A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.~~

~~1.5.2 CHANNEL TEST~~

~~A CHANNEL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practical to verify OPERABILITY, including alarm and/or trip functions.~~

~~1.5.3 CHANNEL CHECK~~

~~A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.~~

~~1.5.4 CHANNEL CALIBRATION~~

~~An instrument CHANNEL CALIBRATION is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.~~

~~1.5.5 HEAT BALANCE CHECK~~

~~A HEAT BALANCE CHECK is a comparison of the indicated neutron power and core thermal power.~~

~~1.5.6 HEAT BALANCE CALIBRATION~~

~~A HEAT BALANCE CALIBRATION is an adjustment of the power range channel amplifiers output based on the core thermal power determination.~~

~~1.6 POWER DISTRIBUTION~~

~~1.6.1 QUADRANT POWER TILT~~

~~Quadrant power tilt is defined by the following equation and is expressed in percent.~~

$$100\left[\frac{\text{Power in Any Core Quadrant}}{\text{Average Power of All Quadrants}} - 1\right]$$

~~The quadrant tilt limits are stated in Specification 3.5.2.4.~~

~~1.6.2 AXIAL POWER IMBALANCE~~

~~Axial power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.~~

~~1.7 CONTAINMENT INTEGRITY~~

~~CONTAINMENT INTEGRITY exists when the following conditions are satisfied:~~

~~a. The equipment hatch is closed and sealed and both doors of the personnel and emergency air locks are closed and sealed.~~

~~b. All passive Containment Isolation Valves (CIVs) and isolation devices, including manual valves and blind flanges, are closed as required by the "Containment Integrity Check List" attached to the operating procedure, "Containment Integrity and Access Limits." Normally closed passive CIVs may be unisolated intermittently under administrative control.~~

~~c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently and manual control of power-operated valves may be substituted for automatic control under administrative control.~~

~~d. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.~~

~~1.8 FIRE SUPPRESSION WATER SYSTEM~~

~~A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.~~

~~1.9 DELETED~~

~~1.10 DELETED~~

~~1.11 DELETED~~

~~1.12 DOSE EQUIVALENT I-131~~

~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same thyroid dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."~~

~~1.13 SOURCE CHECK~~

~~A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.~~

~~1.14 DELETED~~

~~1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)~~

~~The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.3 and 6.9.4.~~

~~1.16 PROCESS CONTROL PROGRAM (PCP)~~

~~The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

~~1.17 GASEOUS RADWASTE TREATMENT~~

~~The GASEOUS RADWASTE TREATMENT SYSTEM is the system designed and installed to reduce radioactive gaseous effluent by collecting primary coolant system off gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

~~1.18 VENTILATION EXHAUST TREATMENT SYSTEM~~

~~A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluent by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodine or particulates from the gaseous exhaust system prior to the release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEMS.~~

~~1.19 PURGE - PURGING~~

~~PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is required to purify the confinement.~~

~~1.20 VENTING~~

~~VENTING is the controlled process of discharging air as gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is not provided. Vent used in system name does not imply a VENTING process.~~

~~1.21 REPORTABLE EVENT~~

~~A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.~~

~~1.22 MEMBER(S) OF THE PUBLIC~~

~~MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the AmerGen Energy Company, LLC, AmerGen Energy Company, LLC contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.~~

~~1.23 SUBSTANTIVE CHANGES~~

~~SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Examples of non substantive changes are: (1) correcting spelling; (2) adding (but not deleting) sign off spaces; (3) blocking in notes, cautions, etc.; (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications; and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.~~

~~1.24 CORE OPERATING LIMITS REPORT~~

~~The CORE OPERATING LIMITS REPORT is a TMI-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these operating limits is addressed in individual specifications.~~

~~1.25 FREQUENCY NOTATION~~

~~The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2. All Surveillance Requirements shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. The 25% extension applies to all frequency intervals with the exception of "F." No extension is allowed for intervals designated "F."~~

~~TABLE 1.2~~

~~FREQUENCY NOTATION~~

~~NOTATION~~	~~FREQUENCY~~
~~S~~	~~Shiftly (once per 12 hours)~~
~~D~~	~~Daily (once per 24 hours)~~
~~W~~	~~Weekly (once per 7 days)~~
~~M~~	~~Monthly (once per 31 days)~~
~~Q~~	~~Quarterly (once per 92 days)~~
~~S/A~~	~~Semi Annually (once per 184 days)~~
~~R~~	~~Refueling Interval (once per 24 months)~~
~~P S/U~~	~~Prior to each reactor startup, if not done during the previous 7 days~~
~~P S/A~~	~~Within six (6) months prior to each reactor startup~~
~~P~~	~~Completed prior to each release~~
~~N/A (NA)~~	~~Not applicable~~
~~E~~	~~Once per 18 months~~
~~F~~	~~Not to exceed 24 months~~

~~1.26 DOSE EQUIVALENT Xe-133~~

~~Dose Equivalent Xe-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT Xe-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12.~~

~~1.27 INSERVICE TESTING PROGRAM~~

~~The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).~~

.

~~Bases~~

~~Section 1.25 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each REFUELING INTERVAL. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each REFUELING INTERVAL. Likewise, it is not the intent that REFUELING INTERVAL surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Section 1.25 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.~~

3/4. LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 GENERAL ACTION REQUIREMENTS AND SURVEILLANCE REQUIREMENT APPLICABILITY

3.0.1 ~~When a Limiting Condition for Operation is not met, except as provided in action called for in the specification, within one hour action shall be initiated to place the unit in a condition in which the specification does not apply by placing it, as applicable, in:~~

~~1. At least HOT STANDBY within the next 6 hours.~~
~~2. At least HOT SHUTDOWN within the following 6 hours, and~~
~~3. At least COLD SHUTDOWN within the subsequent 24 hours.~~

~~Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the time limits of the specification as measured from the time of failure to meet the Limiting Condition for Operation. Applicability of these requirements is stated in the individual specifications.~~

~~Specification 3.0.1 is not applicable in COLD SHUTDOWN OR REFUELING SHUTDOWN.~~LCOs shall be met during the specified conditions in the TS, except as provided in 3.0.2.

3.0.2 Upon discovery of a failure to meet an LCO, the required actions of the associated Conditions shall be met.

If the LCO is met or is no longer applicable prior to expiration of the specified completion time(s), completion of the required action(s) is not required, unless otherwise stated.

4.0.1 ~~During Reactor Operational Conditions for which a Limiting Condition for Operation (LCO) does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/ component operable, the associated surveillance requirement must be current. Failure to perform a surveillance within the specified Frequency shall be failure to meet the LCO except as provided in 4.0.2~~Surveillance requirements shall be met during the specified conditions in the applicability for individual LCOs, unless otherwise stated in the surveillance requirements. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in 4.0.2.

4.0.2 If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. ~~A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.~~

If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

4.0.3 The specified frequency for each SR is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.

BASES

LCO 3.0.1 and LCO 3.0.2, and SR 4.0.1 through SR 4.0.3 ~~This specification~~ delineates the actions to be taken for circumstances not directly provided for in the action requirements of individual specifications and whose occurrence would violate the intent of the specification.

LCO 3.0.1 establishes the applicability statement within each individual specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the applicability statement of each Specification).

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated actions shall be met. The completion time of each required action for an ACTIONS condition is applicable from the point in time that an actions condition is entered. The required actions establish those remedial measures that must be taken within specified completion times when the requirements of an LCO are not met. This specification establishes that completion of the required actions within the specified completion times constitutes compliance with a specification.

Completing the required actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual specifications.

SR 4.0.1 establishes the requirement that SRs must be met during the ~~REACTOR OPERATING CONDITIONS or other~~ specified conditions in the SRs for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification is to ensure that surveillances are performed in order to verify that facility conditions ~~the OPERABILITY of systems and components, and that variables~~ are within specified limits. Failure to meet a surveillance within the specified frequency~~, in accordance with definition 1.25,~~ constitutes a failure to meet an LCO. ~~Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified frequency.~~

Variables ~~Systems and components~~ are assumed to be within limits ~~OPERABLE~~ when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that variables are within limits when the requirements of the surveillance(s) are known not to be met between required surveillance performances.~~systems or components are OPERABLE when:~~

~~a. The system or components are known to be inoperable, although still meeting the SRs or~~

~~b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.~~

Surveillances do not have to be performed when the unit is in a ~~REACTOR OPERATING CONDITION or other~~ specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given ~~REACTOR OPERATING CONDITION or other~~ specified condition.

Surveillances, including surveillances invoked by LCO required actions, do not have to be performed on inoperable equipment because the actions define the remedial measures that apply. Surveillances have to be met and performed in accordance with the specified frequency, prior to returning equipment to OPERABLE status.

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with the specified frequency. Post maintenance testing may not be possible in the current REACTOR OPERATING CONDITION or other specified conditions in the SRs due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a REACTOR OPERATING CONDITION or other specified condition where other necessary post maintenance tests can be completed.~~

~~Some examples of this process are:~~

~~a. Emergency feedwater (EFW) pump maintenance during refueling that requires testing at steam pressures greater than 750 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the EFW pump testing.~~

~~b. High pressure injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.~~

SR 4.0.2 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been performed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the required surveillance has not been performed in accordance with Surveillance ~~Standard~~ Requirement 4.0.2 and not at the time that the specified frequency was not met.

The delay period provides an adequate time to perform surveillances that have been missed. This delay period permits the performance of a surveillance before complying with required actions or other remedial measures that might preclude performance of the surveillance.

The basis for this delay period includes consideration of facility ~~unit~~ conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

~~When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering power operation after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Surveillance Standard 4.0.2 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity. When a Section 6.8, "Procedures and Programs," specification states that the provisions of TS 4.02 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.~~

~~Surveillance Standard 4.0.2 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of operating condition changes imposed by required LCO actions.~~

SR 4.0.2 is only applicable if there is a reasonable expectation the associated ~~equipment is OPERABLE or that~~ variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed. ~~An example of the use of SR 4.0.2 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment.~~ The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

Failure to comply with specified surveillance frequencies is expected to be an infrequent occurrence. Use of the delay period established by Surveillance Standard 4.0.2 is a flexibility which is not intended to be used repeatedly to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. ~~The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. This risk impact should be managed through the programin place to implement 10 CFR 50.65 (a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants'. This~~

~~Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the licensee's Corrective Action Program.~~

If a surveillance is not completed within the allowed delay period, then ~~the equipment is considered inoperable or~~ the variable is considered outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then ~~the equipment is inoperable, or~~ the variable is outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the completion time of the actions, restores compliance.

SR 4.0.3 permits a 25% extension of the interval specified in the frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any Surveillance is the verification of conformance with the SRs.

3/4.1 HANDLING AND STORAGE OF IRRADIATED FUEL IN THE SPENT FUEL POOL

3/4.1.1 SPENT FUEL POOL WATER LEVEL

Applicability

Applies to the minimum level of water in the Spent Fuel Pool during handling of irradiated fuel in the Spent Fuel Pool.

Objective

Ensures that assumptions of Fuel Handling Accident are maintained during handling of irradiated fuel in the Spent Fuel Pool.

Specification

3.1.1.1 Maintain Spent Fuel Pool level greater than 342'4" elevation.

3.1.1.2 With Spent Fuel Pool level less than 342'4" elevation, immediately suspend handling of irradiated fuel in the Spent Fuel Pool.

SURVEILLANCE REQUIREMENTS

4.1.1.1 Verify Spent Fuel Pool level greater than or equal to 342'4" elevation every 7 days.

Bases

The top of fuel is at the 319’4” elevation. The FHA analysis assumes 23’ of water above the fuel assemblies. This dictates a minimum elevation of water in the Spent Fuel Pool of 342’4”. This specification provides the controls to ensure the assumptions of the accident analysis while fuel handling evolutions are in progress. This specification will have a SR 4.1.1.1 that will verify the Spent Fuel Pool water level on a frequency of 7 days.

The water contained in the spent fuel pool provides a medium for removal of decay heat from the stored fuel elements, normally via the spent fuel cooling system. The spent fuel pool water also provides shielding to reduce the general area radiation dose during both spent fuel handling and storage. The resultant 2-hour dose to a person at the exclusion area boundary and the 30-day dose at the low population zone are much less than 10 CFR 50.67 limits.

LCO 3.1.1.2 requires that when the water level in the SFP is lower than the required level, the movement of irradiated fuel assemblies in the SFP is to be "immediately" suspended. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, such that the suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring in the SFP when the level is below the required elevation.

Although maintaining adequate spent fuel pool water level is essential to both decay heat removal and shielding effectiveness, the Technical Specification minimum water level limit is based upon maintaining the pool's iodine retention-effectiveness consistent with that assumed in the evaluation of the Post Permanent Shutdown FHA analysis. The Post Permanent Shutdown FHA analysis assumes that a minimum of 23 feet of water is maintained above the stored fuel. This assumption allows the use of the pool iodine decontamination factor of 200 used in the associated offsite dose calculation.

3/4.1.2 SPENT FUEL POOL BORON CONCENTRATION

Applicability

Applies to the minimum boron concentration in the Spent Fuel Pool during storage and handling of irradiated fuel in the Spent Fuel Pool.

Objective

Ensures that assumptions of Storage Limitations are maintained to prevent inadvertent criticality in the Spent Fuel Pool.

Specification

3.1.2.1 Maintain Spent Fuel Pool boron concentration greater than or equal to 600 ppm.

3.1.2.2 With Spent Fuel Pool boron concentration less than 600 ppm, immediately suspend handling of irradiated fuel in the Spent Fuel Pool and immediately restore boron concentration per 3.1.2.1.

SURVEILLANCE REQUIREMENTS

4.1.2.1 Verify Spent Fuel Pool boron concentration greater than or equal to 600 ppm every 7 days.

Bases

The acceptance criteria for the fuel storage pool criticality analyses is that a keff of < 0.95 must be maintained for all postulated events. The storage racks are capable of maintaining this keff with unborated pool water at a temperature yielding the highest reactivity (assuming the storage restrictions of LCO 3.1.3 are met). Most abnormal storage locations will not result in an increase in the keff of the racks. However, it is possible to postulate events, such as the mis-loading of an assembly with a burnup and enrichment combination outside the acceptable area in Figure 3.1.3-1 and 3.1.3-2, or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. For such events, credit is taken for the presence of boron in the pool water since the NRC does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in keff, caused by the boron more than offsets the reactivity addition caused by credible accidents. This specification will have a Surveillance Requirement SR 4.1.2.1 that will verify the Spent Fuel Pool Boron on a frequency of 7 days.

LCO 3.1.2.2 requires that when the SFP boron concentration is less than 600 ppm, the movement of irradiated fuel assemblies in the SFP is to be "immediately" suspended. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, such that the suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring in the SFP when the boron concentration is below the required level.

3/4.1.3 SPENT FUEL ASSEMBLY STORAGE

Applicability

Applies whenever any fuel assembly is stored in Storage Pool A or Storage Pool B of the Spent Fuel Pool.

Objective

Ensures that assumptions of Storage Limitations are maintained to prevent inadvertent criticality in the Spent Fuel Pool.

Specification

3.1.3.1 The combination of initial enrichment and burnup of each spent fuel assembly stored in Storage Pool A and Storage Pool B, shall be within the acceptable region of Figure 3.1.3-1 or 3.1.3-2.

3.1.3.2 When requirement of 3.1.3.1 is not met, immediately initiate action to move the noncomplying fuel assembly to an acceptable configuration.

SURVEILLANCE REQUIREMENTS

4.1.3.1 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.1.3-1 or Figure 3.1.3-2 prior to storing irradiated spent fuel in the Spent Fuel Pool A or Spent Fuel Pool B.

Bases

The function of the spent fuel storage racks is to support safety analyses and protect spent fuel assemblies from the time they are placed in the pool until they are shipped offsite. The spent fuel assembly storage LCO was derived from the need to establish limiting conditions on fuel storage to assure sufficient safety margin exists to prevent inadvertent criticality. The spent fuel assemblies are stored entirely underwater in a configuration that has been shown to result in a reactivity of less than or equal to 0.95 under worse case conditions. The spent fuel assembly enrichment requirements in this LCO are required to ensure inadvertent criticality does not occur in the spent fuel pool. Inadvertent criticality within the fuel storage area could result in offsite radiation doses exceeding 10 CFR 50.67 limits.

LCO 3.1.3.2 requires that when LCO 3.1.3.1 is not met, "immediately" initiate action to move the noncomplying fuel assembly to an acceptable configuration. "Immediately" as used in this completion time means the required action should be pursued without delay and in a controlled manner, to reestablish the safety margins to prevent an inadvertent criticality.

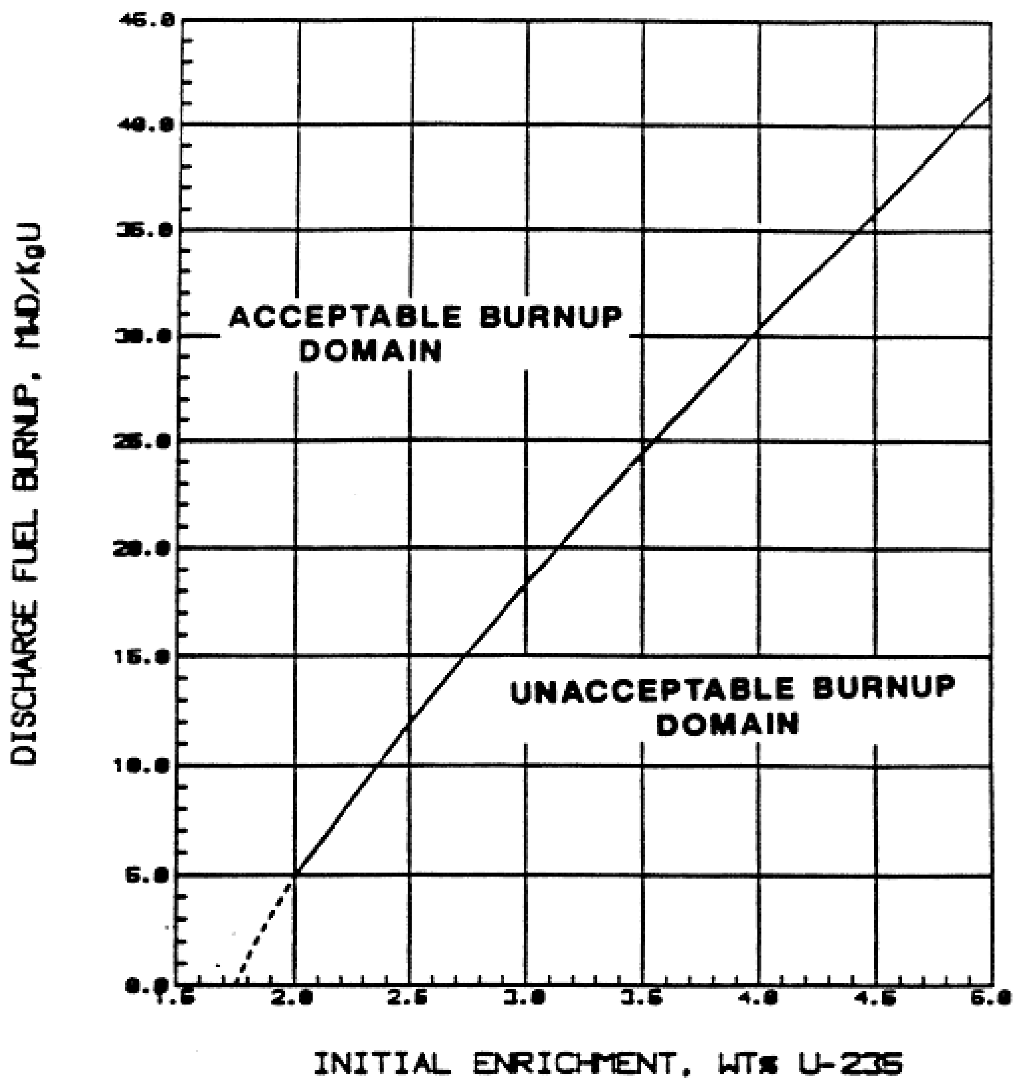


Figure 3.1.3-1
Minimum Burnup Requirements for Fuel in Region II of the Pool A Storage Racks

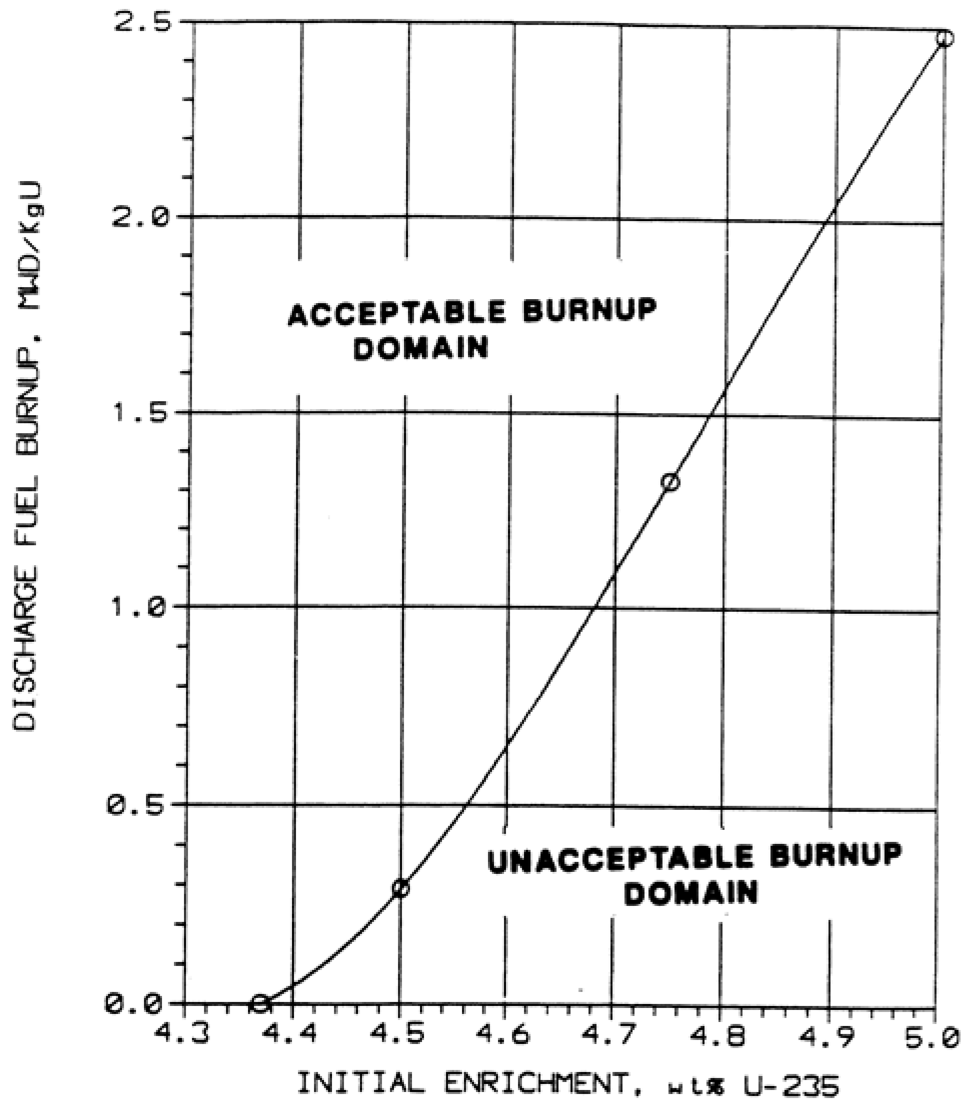


Figure 3.1.3-2
Minimum Burnup Requirements for Fuel in the Pool B Storage Racks

3/4.1.4~~1~~ HANDLING OF IRRADIATED FUEL WITH THE FUEL HANDLING BUILDING CRANE

Applicability

Applies to the operation of the fuel handling building crane when within the confines of Unit 1 and there is any spent fuel in storage in the Unit 1 fuel handling building.

Objective

To define the lift conditions and allowable areas of travel when loads to be lifted and transported with the fuel handling building crane are in excess of 15 tons or between 1.5 tons and 15 tons or consist of irradiated fuel elements.

Specification

3.~~11~~1.4.1 Spent fuel elements having less than 120 days for decay of their irradiated fuel shall not be loaded into a spent fuel transfer cask in the shipping cask area.

3.~~11~~1.4.2 The key operated travel interlock system for automatically limiting the travel area of the fuel handling building crane shall be imposed whenever loads in excess of 15 tons are to be lifted and transported with the exception of fuel handling bridge maintenance.

3.~~11~~1.4.3 The lowest surface of all loads in excess of 15 tons shall be administratively limited to an elevation one foot or less above the concrete surface at the nominal 348 ft-0 in. elevation in the fuel handling building.

3.~~11~~1.4.4 Loads in excess of hook capacity shall not be lifted, except for load testing.

3.~~11~~1.4.5 Following modifications or repairs to any of the load bearing members, the crane shall be subjected to a test lift of 125 percent of its rated load.

3.~~11~~1.4.6 Administrative controls shall require the use of an approved procedure with an identified safe load path for loads in excess of 3,000 lbs. handled above the Spent Fuel Pool Operating Floor (348' elevation).

3.~~11~~1.4.7 During transfer of the cask to and from the cask loading pit, the cask will be restricted to the transfer path shown in Figure 3.11-1. Administrative controls will be used to ensure that all lateral movements of the cask are performed at slow bridge and trolley speeds. During this transfer the cask lifting yoke shall be oriented in the East-West direction.

Bases

This specification will limit activity releases to unrestricted areas resulting from damage to spent fuel stored in the spent fuel storage pools in the postulated event of the dropping of a heavy load from the fuel handling building crane. A Fuel Handling accident analysis was performed assuming that the cask and its entire contents of ten fuel assemblies are sufficiently damaged as a result of dropping the cask, to allow the escape of all noble gases and iodine in the gap (Reference 1). This release was assumed to be directly to the atmosphere and to occur instantaneously. The site boundary doses resulting from this accident are 5.25 R whole body and 1.02 R to thyroid, and are within the limits specified in 10 CFR 100.

Specification 3.~~11~~1.4.1 requires that spent fuel, having less than 120 days decay post-irradiation, not be loaded in a spent fuel transfer cask in order to ensure that the doses resulting from a highly improbable spent fuel transfer cask drop would be within those calculated above.

Specification 3.~~11~~1.4.2 requires the key operated interlock system, which automatically limits the travel area of the fuel handling crane while it is lifting and transporting the spent fuel shipping cask, to be imposed whenever loads in excess of 15 tons are to be lifted and transported while there is any spent fuel in storage in the spent fuel storage pools in Unit 1. This automatically ensures that these heavy loads travel in areas where, in the unlikely event of a load drop accident, there would be no possibility of this event resulting in any damage to the spent fuel stored in the pools, any unacceptable structural damage to the spent fuel pool structure, or damage to redundant trains of safety related components. The shipping cask area is designed to withstand the drop of the spent fuel shipping cask from the 349 ft-0 in. elevation without unacceptable damage to the spent fuel pool structure (Reference 2).

Specification 3.~~11~~1.4.3 ensures that the lowest surface of any heavy load never gets higher than one foot above the concrete surface of the 348 ft-0 in. elevation in the fuel handling building (nominal elevation 349 ft-0 in.) thereby keeping any impact force from an unlikely load drop accident within acceptable limits.

Specification 3.~~11~~1.4.4 ensures that the proper capacity crane hook is used for lifting and transporting loads thus reducing the probability of a load drop accident.

Following modification or repairs, specification 3.~~11~~1.4.5 confirms the load rating of the crane.

Specification 3.~~11~~1.4.6 imposes administrative limits on handling loads weighing in excess of 3000 lbs. to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the spent fuel pool, or to impact redundant safe shutdown equipment. The safe load path shall follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. Handling loads of less than 3000 lbs. without these restrictions is acceptable because the consequences of dropping loads in this weight range are comparable to those produced by the fuel handling accident considered in the FSAR and found acceptable.

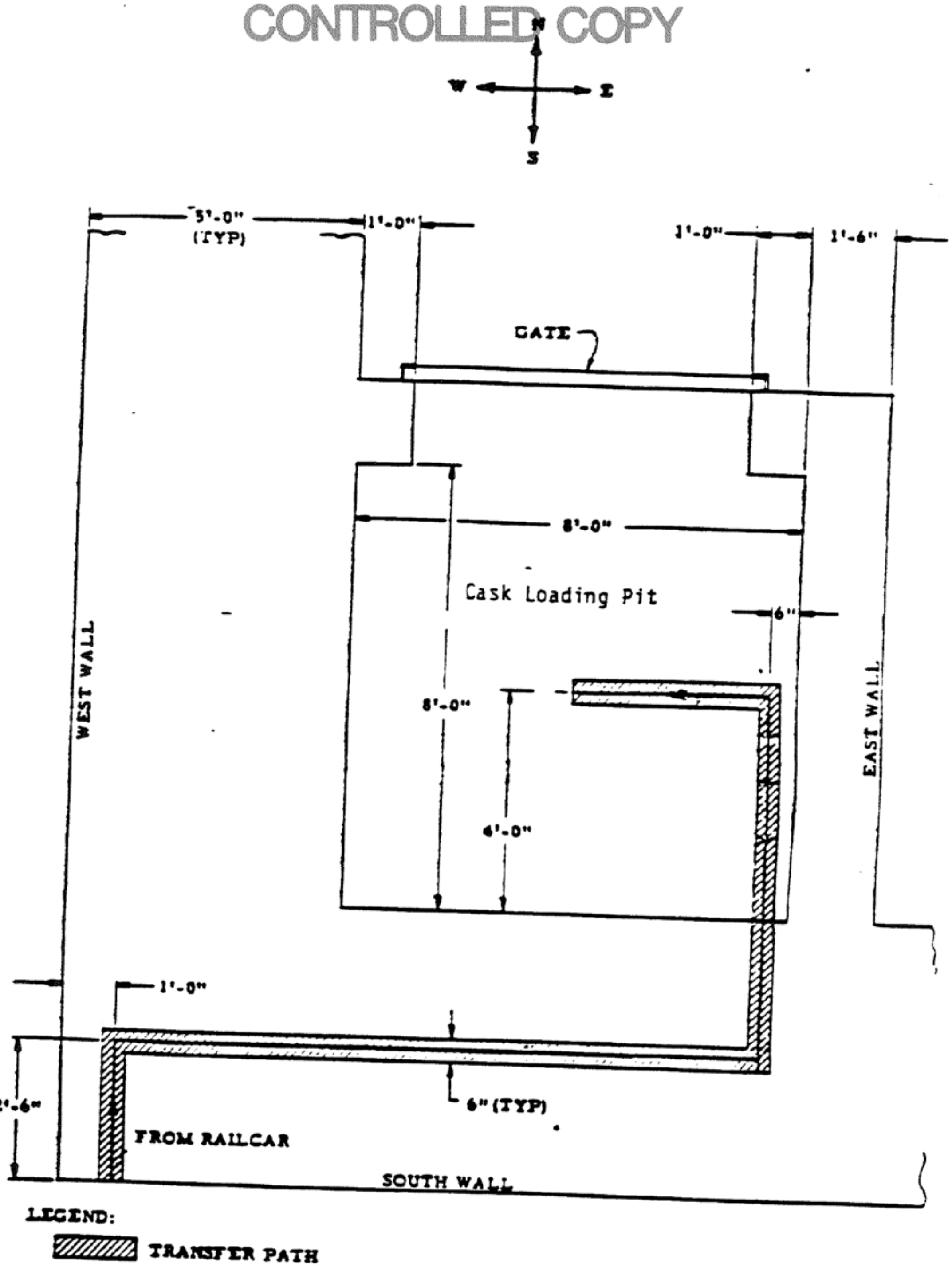
Specification 3.~~11~~1.4.7 in combination with 3.~~11~~1.4.3 ensures the spent fuel cask is handled in a manner consistent with the load drop analysis (Reference 3).

References

(1) UFSAR, Section 14.2.2.1 - “Fuel Handling Accident”

(2) UFSAR, Section 14.2.2.8 - “Fuel Cask Drop Accident”

(3) GPU Evaluation of Heavy Load Handling Operations at TMI-1 February 21, 1984, as transmitted to the NRC in GPUN Letter No. 5211 84 2013.



TRANSFER PATH TO AND FROM CASK LOADING PIT
(EL. 348'-0")

FIGURE 3.11-1

5.0 DESIGN FEATURES

5.1 SITE

Applicability

Applies to the location and extent of the exclusion boundary, restricted area, and low population zone.

Objective

To define the above by location and distance description.

Specification

5.1.1 The Three Mile Island Nuclear Station Unit 1 is located in an area of low population density about ten miles southeast of Harrisburg, PA. It is in Londonderry Township of Dauphin County, Pennsylvania, about two and one-half miles north of the southern tip of Dauphin County, where Dauphin is coterminal with York and Lancaster Counties. The station is located on an island approximately three miles in length situated in the Susquehanna River upstream from York Haven Dam. Figure 5-1 is an extended plot plan of the site showing the plant orientation and immediate surroundings. The description of the Exclusion Area as defined in 10 CFR 100.3, is located in the Final Safety Analysis Report, as updated.~~a 2,000 ft. radius, including portions of Three Mile Island, the river surface around it, and a portion of Shelley Island, which is owned by Exelon Generation Company, LLC. The minimum distance of 2,000 ft. occurs on the shore of the mainland in a due easterly direction from the plant as shown on Figure 5-1 for the Exclusion Area. Figure 5-3 showing the physical location of the fence defines the "Restricted Area" surrounding the plant. The minimum distance of the "Restricted Area" is approximately 560 feet and is from the centerline of the TMI Unit 2 Reactor Building to a point on the westerly shoreline of Three Mile Island. The minimum distance to the outer boundary of the low population zone is two miles as shown on T.S. Figure 5-2, which also depicts the site topography for a radius of five miles. T.S. Figure 5-3 depicts the locations of gaseous effluent release points and liquid effluent outfalls (as tabularized on page 5-10), and the meteorological tower location (designated as 'weather tower' on the figure).~~

~~5.2 CONTAINMENT~~

~~Applicability~~

~~Applies to those design features of the containment system relating to operational and public safety.~~

~~Objective~~

~~To define the significant design features of the reactor containment.~~

~~Specification~~

~~Containment consists of two systems which are the reactor building and reactor building isolation system.~~

~~5.2.1 REACTOR BUILDING~~

~~The reactor building completely encloses the reactor and the associated reactor coolant systems. The reactor building is a reinforced concrete structure composed of cylindrical walls with a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild-steel reinforcing. The cylindrical walls are prestressed with a post-tensioning tendon system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three-way post-tensioning tendon system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness for containment.~~

~~The internal free volume of the reactor building is in excess of 2.0x106 cubic feet. The foundation mat is 9 ft thick with a 2 ft thick concrete slab above the bottom liner plate. The cylindrical portion has an inside diameter of 130 ft, wall thickness of 3 ft 6 in., and a height of 157 ft from top of foundation slab to the spring line. The shallow dome roof has a large radius of 110 ft, a transition radius of 20 ft 6 in., a thickness of 3 ft, and an overall height of 32 ft 4 1/8 in.~~

~~The concrete containment building provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 55 psig and 281°F, respectively. The reactor building is designed for an external atmospheric pressure of 2.5 psi greater than the internal pressure.~~

~~Penetration assemblies are welded to the reactor building liner. Access openings, electrical penetrations, and fuel transfer tube covers are equipped with double seals. Reactor building purge penetrations and reactor building atmosphere sampling penetrations are equipped with double valves having resilient seating surfaces (Reference 1).~~

~~The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in Section 14, with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safeguards, and the combined influence of the energy sources and heat sinks.~~

~~5.2.2 REACTOR BUILDING ISOLATION SYSTEM~~

~~Leakage through all fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves (Reference 2).~~

~~REFERENCES~~

~~(1) UFSAR Section 5.2.2.4.8 - "Penetrations and Openings"~~

~~(2) UFSAR Section 5.3.1 - "Isolation System - Design Bases"~~

~~5.3 REACTOR~~

~~Applicability~~

~~Applies to the design features of the reactor core and reactor coolant system.~~

~~Objective~~

~~To define the significant design features of the reactor core and reactor coolant system.~~

~~Specification~~

~~5.3.1 REACTOR CORE~~

~~5.3.1.1 A fuel assembly normally contains 208 fuel rods arranged in a 15 by 15 lattice. The reactor shall contain 177 fuel assemblies. Fuel rods shall be clad with zircaloy, ZIRLO, or zirconium-based M5 alloy materials and contain an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. The details of the fuel assembly design are described in TMI-1 UFSAR Chapter 3.~~

~~5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches. The active fuel height is defined in TMI-1 UFSAR Chapter 3.~~

~~5.3.1.3 The core average and individual batch enrichments for the present cycle are described in TMI-1 UFSAR Chapter 3.~~

~~5.3.1.4 The control rod assemblies (CRA) are distributed in the reactor core as shown in TMI-1 FSAR Chapter 3. The CRA design data are also described in the UFSAR.~~

~~5.3.1.5 The TMI-1 core may contain burnable poison rod assemblies (BPRA) and gadolinia-urania integral burnable poison fuel pellets as described in TMI-1 UFSAR Chapter 3.~~

~~5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation data described in the UFSAR. Enrichment shall not exceed a nominal 5.0 weight percent of U235.~~

~~5.3.2 REACTOR COOLANT SYSTEM~~

~~5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements. (Refer to UFSAR Chapter 4 for details of design and operation.)~~

~~5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F.~~

~~5.3.2.3 The reactor coolant system volume shall be less than 12,200 cubic feet.~~

5.~~4~~2 ~~NEW AND~~ SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for ~~new and~~ spent fuel assemblies.

Objective

To assure that ~~both new and~~ spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.2~~4~~.1 ~~NEW~~ SPENT FUEL STORAGE

~~a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools.~~

~~For the new fuel storage vault, the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions. The spacing in the new fuel storage vault is sufficient to maintain Keff less than 0.95 based on storage of fuel assemblies in clean unborated water or less than 0.98 based on storage in an optimum hypothetical low density moderator (fog or foam) for fuel assemblies with a nominal enrichment of 5.0 weight percent U235. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition.~~

For Spent Fuel Pool "A", the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 11.1 inches in both directions for the Region I racks and 9.2 inches in both directions for the Region II racks. The spacing in the Spent Fuel Pool "A" storage locations for both Region I and II is adequate to maintain Keff less than 0.95. Region I will store fuel with a maximum 5.0 percent initial enrichment. ~~Region II will store new fuel with low enrichment.~~ When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppm~~b~~ must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition.

For Spent Fuel Pool "B", the fuel assemblies are stored in racks in parallel rows, having nominal center to center distance of 13-5/8 inches in both directions. This spacing is sufficient to maintain a Keff less than 0.95 based on fuel assemblies with a maximum enrichment of 4.37 weight percent U235. When fuel is being moved in or over the Spent Fuel Storage Pool "B" and fuel is being stored in the pool, a boron concentration of at least 600 ppm~~b~~ must be maintained meet the NRC maximum allowable reactivity value under the postulated accident condition.

~~b. Deleted.~~

~~c. New fuel may also be stored in shipping containers.~~

~~5.4.2 SPENT FUEL STORAGE (Reference 1)~~

a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.

b. ~~Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.~~

~~c. Deleted.~~

~~d.~~ The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:

	Spent Fuel Pool A North End of Fuel Handling Building	Spent Fuel Pool B South End of Fuel Handling Building	Dry New Fuel Storage Area Fuel Handling Building
Fuel Assys.	1494 *	496	54
Cores	8.44	2.8	0.37

NOTE: * Includes three spaces for accommodating failed fuel containers.

e. All of the fuel assembly storage racks provided are designed to Seismic Class 1 criteria to the accelerations indicated below:

	Fuel Handling Building Dry New Fuel Storage Area And Spent Fuel Pool A	Fuel Handling Building Spent Fuel Pool B
Horiz.	0.38 g	**
Vertical	0.25 g	**

NOTE: ** The "B" pool fuel storage racks are designed using the floor response spectra of the Fuel Handling Building.

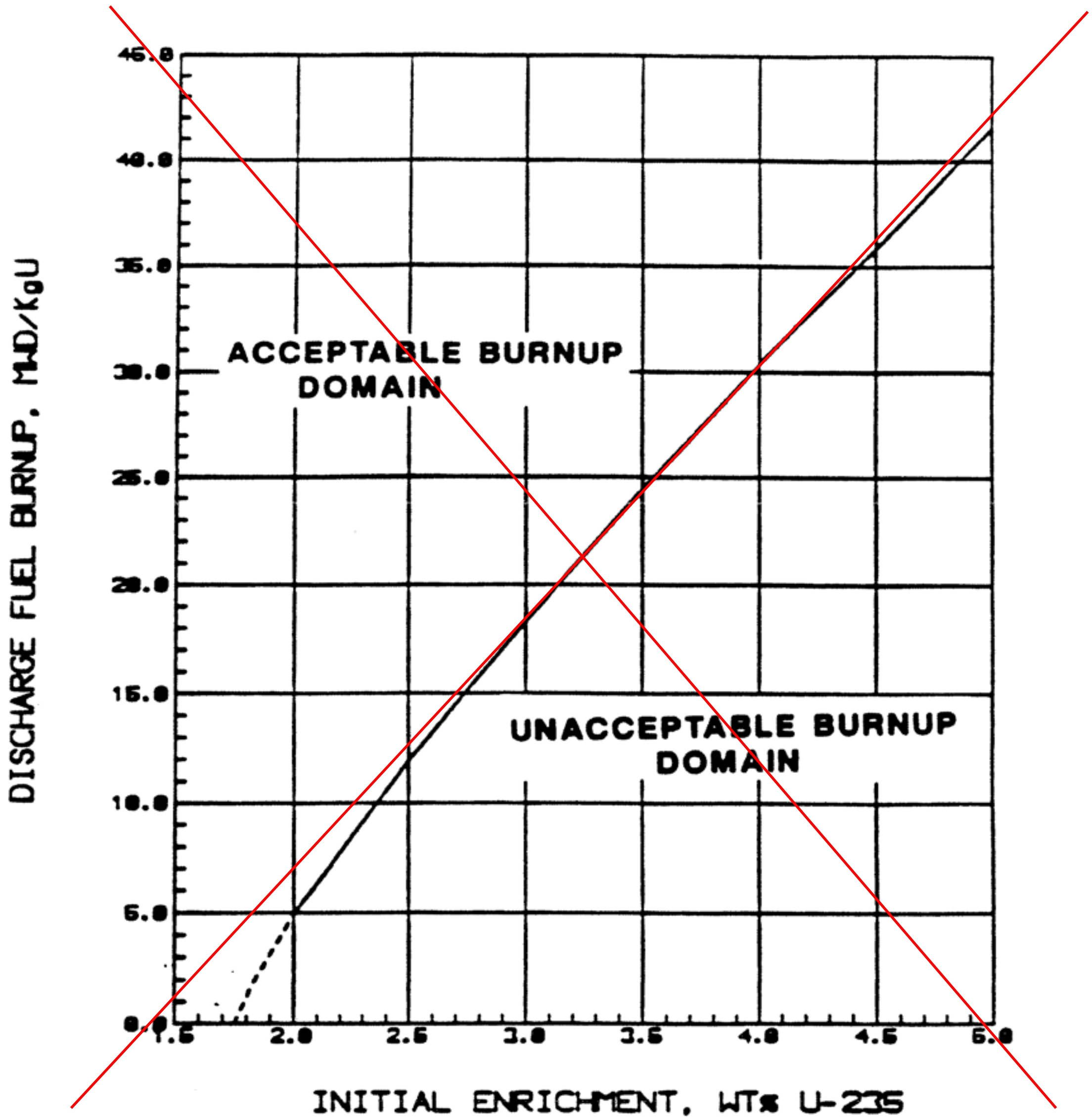
~~f. DELETED~~

~~g. When spent fuel assemblies are stored in the Spent Fuel Pool "A", Region II storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-4.~~

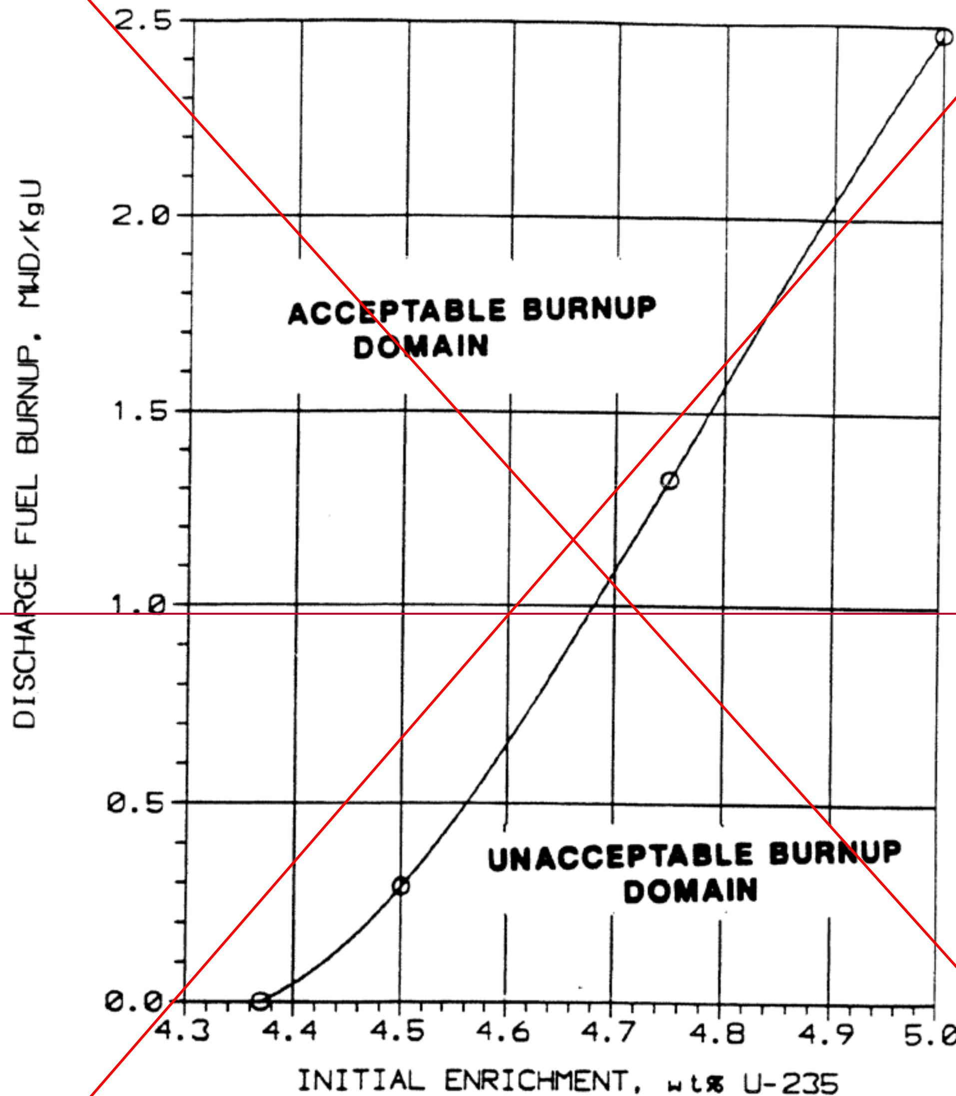
~~h. When spent fuel assemblies are stored in the Spent Fuel Pool "B", storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-5.~~

REFERENCES

(1) UFSAR, Section 9.7 - "Fuel Handling System"



~~Figure 5-4 MINIMUM BURNUP REQUIREMENTS FOR FUEL IN REGION II OF THE POOL A STORAGE RACKS~~



~~Figure 5-5 MINIMUM BURNUP REQUIREMENTS FOR FUEL IN THE POOL "B" STORAGE RACKS~~

6.8.4 a. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

(1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

~~(2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~

~~(3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

b. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public ~~MEMBERS OF THE PUBLIC~~ from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall
be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

(1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

(2) Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 times the concentrations specified in 10 CFR Part 20.1001 - 20.2402, Appendix B, Table 2, Column 2,

(3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

b. Radioactive Effluent Controls Program (continued)

(4) Limitations on the annual and quarterly doses or dose commitment to a member of the public ~~MEMBER OF THE PUBLIC~~ from radioactive materials in liquid effluents released from the unit to the site boundary conforming to Appendix I to 10 CFR Part 50,

(5) Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

(6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

(7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at, or beyond, the site boundary. The limits are as follows:

(a) For noble gases: less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

(b) For I-131, I-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: less than or equal to 1500 mrem/yr to any organ.

(8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,

(9) Limitations on the annual quarterly doses to a member of the public~~MEMBER OF THE PUBLIC~~ from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50, and

(10) Limitations on the annual dose or dose commitment to any member of the public ~~MEMBER OF THE PUBLIC~~ due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

~~6.8.5~~ ~~Reactor Building Leakage Rate Testing Program~~

~~The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:~~

~~A program shall be established to implement the leakage rate testing of the Reactor Building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":~~

~~a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed prior to startup form the T1R18 refueling outage. The T1R18 refueling outage will begin no later than November 1, 2009.~~

~~The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident,~~ $\cancel{P_{ac}}$~~, is 50.6 psig.~~

~~The maximum allowable Reactor Building leakage rate,~~ $\cancel{L_a}$~~, shall be 0.1 weight percent of containment atmosphere per 24 hours at~~ $\cancel{P_{ac}}$~~.~~

~~Reactor Building leakage rate acceptance criteria is ≤ 1.0~~ $\cancel{L_a}$~~. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60~~ $\cancel{L_a}$ ~~for the Type B and Type C tests and ≤ 0.75~~ $\cancel{L_a}$ ~~for the Type A tests.~~

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the NRC Region 1 Office unless otherwise noted.

6.9.1 Routine Reports

A. ~~DELETED.~~

~~B.~~ Annual Reports. Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year. (A single submittal maybe made for the station. The submittal should combine those sections that are common to both units at the station.)

1. ~~DELETED~~

~~2.~~ The following information on aircraft movements at the Harrisburg International Airport:

a. The total number of aircraft's movements (takeoffs and landings) at the Harrisburg International Airport for the previous twelve-month period.

b. The total number of movements of aircraft larger than 200,000 pounds at the Harrisburg International Airport for the previous twelve-month period, broken down into scheduled and non-scheduled (including military) takeoffs and landings, based on a current estimate provided by the airport manager or his designee.

~~3. DELETED~~

~~4. DELETED~~

~~5. DELETED~~

~~C. DELETED~~

~~6.9.2 DELETED~~

6.9.2~~3~~ ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.2~~3~~.1 The Annual Radiological Environmental Operating Report covering the ~~operation of the unit~~ facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in: (1) the ODCM; and, (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

Note: A single submittal may be made for the station.

6.9.34 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.34.1 The Annual Radioactive Effluent Release Report covering the ~~operation of the unit~~facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include a summary of the quantities of radioactive liquid and gaseous effluent and solid waste released from the unit. The material provided shall be: (1) consistent with the objectives outlined in the ODCM and PCP; and, (2) in conformance with 10 CFR 50.36(a) and Section IV.B.1 of Appendix I to 10 CFR Part 50.

Note: A single submittal may be made for the station. The submittal should combine those sections that are common to both units at the station.

~~6.9.5 CORE OPERATING LIMITS REPORT~~

~~6.9.5.1 The core operating limits addressed by the individual Technical Specifications shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle.~~

~~6.9.5.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at TMI-1, specifically:~~

~~(1) BAW-10179 P-A, "Safety and Methodology for Acceptable Cycle Reload Analyses." The current revision level shall be specified in the COLR.~~

~~(2) TR-078-A, "TMI-1 Transient Analyses Using the RETRAN Computer Code", Revision 0. NRC SER dated 2/10/97.~~

~~(3) TR-087-A, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code", Revision 0. NRC SER dated 12/19/96.~~

~~(4) TR-091-A, "Steady State Reactor Physics Methodology for TMI-1", Revision 0. NRC SER dated 2/21/96.~~

~~(5) TR-092P-A, "TMI-1 Reload Design and Setpoint Methodology", Revision 0. NRC SER dated 4/22/97.~~

~~(6)~~

~~(6) BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", NRC SER dated February 4, 2000.~~

~~6.9.5.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient/accident analysis limits) of the safety analysis are met.~~

~~6.9.5.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~

~~6.9.6 STEAM GENERATOR TUBE INSPECTION REPORT~~

~~A report shall be submitted within 180 days after the average reactor coolant temperature exceeds 200°F following completion of an inspection performed in accordance with Section 6.19, Steam Generator (SG) Program. The report shall include:~~

~~a. The scope of inspections performed on each SG,~~

~~b. Degradation mechanisms found,~~

~~c. Nondestructive examination techniques utilized for each degradation mechanism,~~

~~d.~~

~~d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,~~

~~e. Number of tubes plugged during the inspection outage for each degradation mechanism,~~

~~f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,~~

~~g. The results of condition monitoring, including the results of tube pulls and in-situ testing.~~

6.10 RECORD RETENTION

6.10.1 Records shall be retained as described by the Decommissioning Quality Assurance Program.~~The following records shall be retained for at least five years:~~

~~a. Records of normal station operation including power levels and periods of operation at each power level.~~

~~b. Records of principal maintenance activities, including inspection, repairs, substitution, or replacement of principal items of equipment related to nuclear safety.~~

~~c. All REPORTABLE EVENTS.~~

~~d. Records of periodic checks, tests and calibrations.~~

~~e. Records of reactor physics tests and other special tests related to nuclear safety.~~

~~f. Changes to procedures required by Specification 6.8.1.~~

~~g. Deleted~~

~~h. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.~~

~~i. Results of annual physical inventory verifying accountability of licensed sources on record.~~

~~j. Control Room Log Book.~~

~~k. Control Room Supervisor Log Book.~~

~~6.10.2 The following records shall be retained for the duration of Operating License DPR-50 unless otherwise specified in 6.10.1 above.~~

~~a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.~~

~~b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.~~

~~c. Routine unit radiation surveys and monitoring records.~~

~~d. Records of doses received by all individuals for whom monitoring was required.~~

~~e. Records of radioactive liquid and gaseous wastes released to the environment, and records of environmental monitoring surveys.~~

~~f. Records of transient or operational cycles for those facility components which affect nuclear safety for a limited number of transients or cycles as defined in the Final Safety Analysis Report.~~

~~g. Records of training and qualification for current members of the unit staff.~~

~~h. Records of in-service inspections performed pursuant to these Technical Specifications.~~

~~i. Records of Quality Assurance activities required by the QATR.~~

~~j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.~~

~~k. Deleted.~~

~~l. Records of analyses required by the radiological environmental monitoring program.~~

~~m. Records of the service lives of all safety related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.~~

~~n. Records of solid radioactive shipments.~~

~~o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.~~

6.11 ~~RADIATION PROTECTION PROGRAM~~DELETED

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation at 30 cm (11.8 in.) is greater than 100 mrem/hr. deep dose but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and personnel desiring entrance shall obtain a Radiation Work Permit (RWP). Any individual or group of individuals entering a High Radiation Area shall (a) use a continuously indicating dose rate monitoring device or (b) use a radiation dose rate integrating device which alarms at a pre-set dose level (entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them), or (c) assure that a radiological control technician provides positive control over activities within the area and periodic radiation surveillance with a dose rate monitoring instrument.

b. In addition to the requirements of specification 6.12.1.a:

1. Any area accessible to personnel where an individual could receive in any one hour a deep dose in excess of 1000 mrem at 30 cm (11.8 in.) but less than 500 rads at one meter (3.28 ft), from sources of radioactivity shall be locked or guarded to prevent unauthorized entry. The keys to these locked barricades shall be maintained under the administrative control of the respective Radiological Controls Supervisor.

2 For individual high radiation areas where an individual could receive in any one hour deep dose in excess of 1000 mrem at 30 cm (11.8 in.) but less than 500 rads at one meter (3.28 ft.), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

The Radiation Work Permit is not required by Radiological Controls personnel during the performance of their assigned radiation protection duties provided they are following radiological control procedures for entry into High Radiation Areas.

6.13 ~~PROCESS CONTROL PROGRAM (PCP)~~DELETED

~~6.13.1 Licensee initiated changes to the PCP:~~

~~1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:~~

~~a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;~~

~~b. a determination that the changes did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and,~~

~~c. documentation that the changes have been reviewed and approved pursuant to 6.8.2.~~

~~2. Shall become effective upon review and approval by licensee management.~~

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Licensee initiated changes to the ODCM:

1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

   a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

   b. a determination that the changes did not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation that the changes have been reviewed and approved pursuant to 6.8.2.

2. Shall become effective upon review and approval by licensee management.

6.15 DELETED

6.16 DELETED

6.17 DELETED

6.18 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

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

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

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

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

2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of Specification 6.18.b.1 or 6.18.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71 (e).

~~6.19 STEAM GENERATOR (SG) PROGRAM~~

~~A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:~~

~~a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.~~

~~b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.~~

~~1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.~~

~~3. The operational leakage performance criterion is specified in TS 3.1.6, "LEAKAGE."~~

~~Provisions for SG tube plugging criteria.~~

~~1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.~~

~~d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

~~1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.~~

~~2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.~~

~~a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;~~
~~b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;~~
~~c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and~~
~~d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.~~

~~3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.~~

~~e. Provisions for monitoring operational primary to secondary leakage.~~

~~NOTE: Refer to Section 6.9.6 for reporting requirements for periodic SG tube inspections.~~

~~6.20 Control Room Envelope Habitability Program~~

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

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

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

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

~~d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the Control Room Ventilation System, operating at the design flow rate, at a Frequency of 24 months. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.~~

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

~~f. The provisions of Section 1.25 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.~~

~~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.~~