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

August 29, 2019

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 – ISSUANCE OF AMENDMENT NO. 297 RE: DEFUELED TECHNICAL SPECIFICATIONS AND REVISED LICENSE CONDITIONS (EPID L-2018-LLA-0204)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 297 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1, in response to your application dated July 25, 2018, as supplemented by letter dated March 6, 2019.

The amendment revises the renewed facility operating license and the associated technical specifications to permanently defueled technical specifications, consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The amendment also changes the current licensing basis mitigation strategies for flood mitigation and aircraft impact protection in the air intake tunnel.

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

Sincerely,

Justin C. Poole, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures:
1. Amendment No. 297 to DPR-50
2. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 297
Renewed License No. DPR-50

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated July 25, 2018, as supplemented by letter dated March 6, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 1.b, 1.c, 1.d, 1.g, 1.h, 2., 2.a, 2.b.(1), 2.b.(2), 2.b.(4), 2.c.(1), 2.c.(2), 2.c.(4), 2.c.(5), 2.c.(18), 2.c.(19), 2.c.(20), 2.c.(22), and 2.d of Renewed Facility Operating License No. DPR-50 are hereby amended to read as follows:

1.b. DELETED

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

1.d. There is a reasonable assurance: (1) that the activities authorized by this renewed 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;

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

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 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;

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

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 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.

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 the facility as required for fuel storage in accordance with the procedures and limitations set forth in this renewed license;

2.b.(2) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to possess 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 in amounts as required;

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 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.

2.c.(1) DELETED

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby incorporated in the license. The Exelon Generation Company shall maintain the facility in accordance with the Permanently Defueled Technical Specifications (PDTS).

2.c.(4) DELETED

2.c.(5) DELETED

2.c.(18) DELETED

2.c.(19) DELETED

2.c.(20) DELETED

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.

2.d. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

3. This license amendment is effective following the docketing of the certifications required by 10 CFR 50.82(a)(1)(i) and (ii) that Three Mile Island Nuclear Station, Unit 1, has been permanently shut down and defueled. The amendment shall be implemented within 30 days of the effective date of the amendment, but will not exceed December 31, 2019.

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: August 29, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 297

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Renewed Facility Operating License with the revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1	1
2	2
3	3
4	4
5	5
6	6
7	---

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

Remove	Insert
i through vii	i through ii
1-1 through 2-12	1-1
3-1 through 4-87	3/4-1 through 3/4-13
5-1 through 5-10	5-1 through 5-3
6-11a through 6-30	6-11a through 6-15

EXELON GENERATION COMPANY, LLC

(Three Mile Island Nuclear Station, Unit 1)

DOCKET NO. 50-289

RENEWED FACILITY 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

   c. The facility will be maintained 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 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 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 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 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 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 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 possess 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 in amounts as required;

      (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 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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby incorporated in the license. The Exelon Generation Company shall maintain the facility in accordance with the Permanently Defueled Technical Specifications (PDTS).

(3) Physical Protection

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

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.

(4) DELETED

(5) DELETED

(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

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

(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

(19) DELETED

(20) DELETED

(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.

(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 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

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LIST OF FIGURES

1. DEFINITIONS

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

1.1 ACTIONS

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

1.2 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.

1.4 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).

1.5 STATION, UNIT, PLANT, AND FACILITY

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

3/4. LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 GENERAL ACTION REQUIREMENTS AND SURVEILLANCE REQUIREMENT APPLICABILITY

3.0.1 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 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. 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 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 delineate 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 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 are within specified limits. Failure to meet a surveillance within the specified frequency constitutes a failure to meet an LCO.

Variables are assumed to be within limits 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.

Surveillances do not have to be performed when the unit is in a 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 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.

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 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 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.

SR 4.0.2 is only applicable if there is a reasonable expectation the associated 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. 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. If a surveillance is not completed within the allowed delay period, then 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 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.

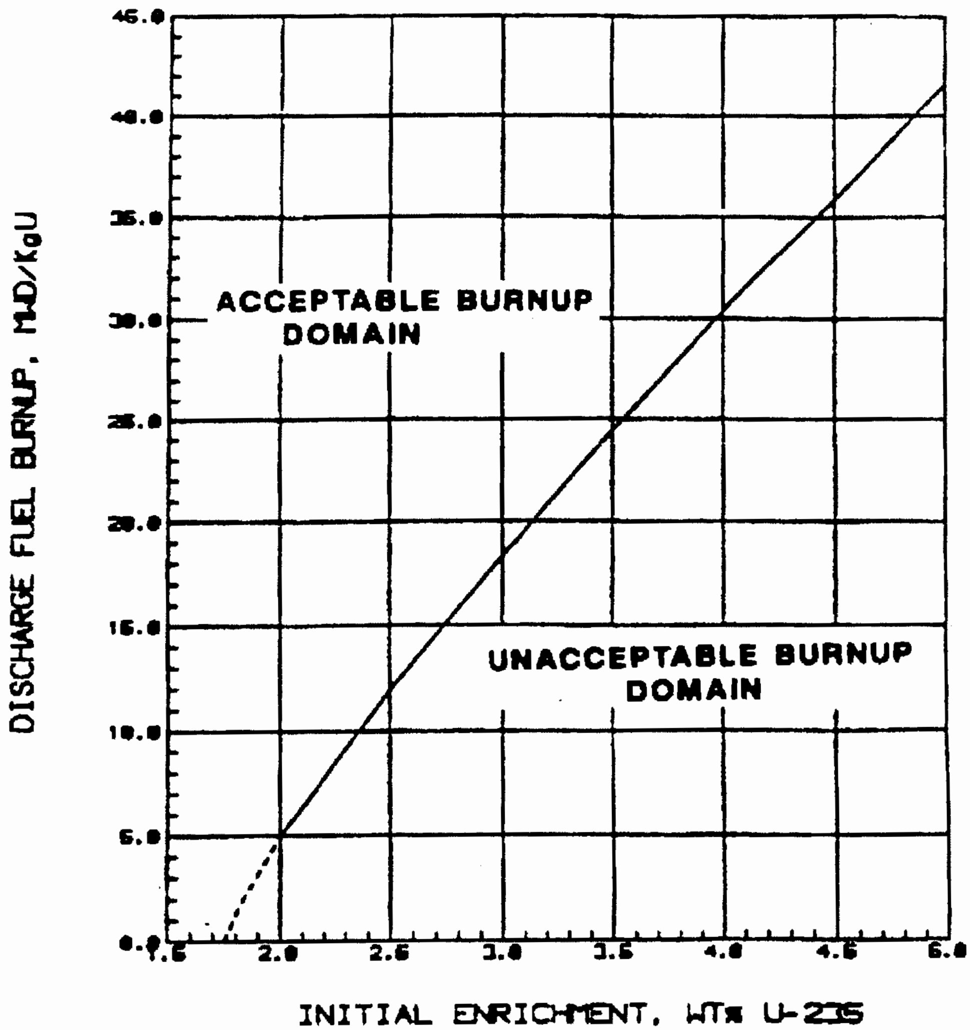

Figure 3.1.3-1
Minimum Burnup Requirements for Fuel in Region II of the Pool A Storage Racks

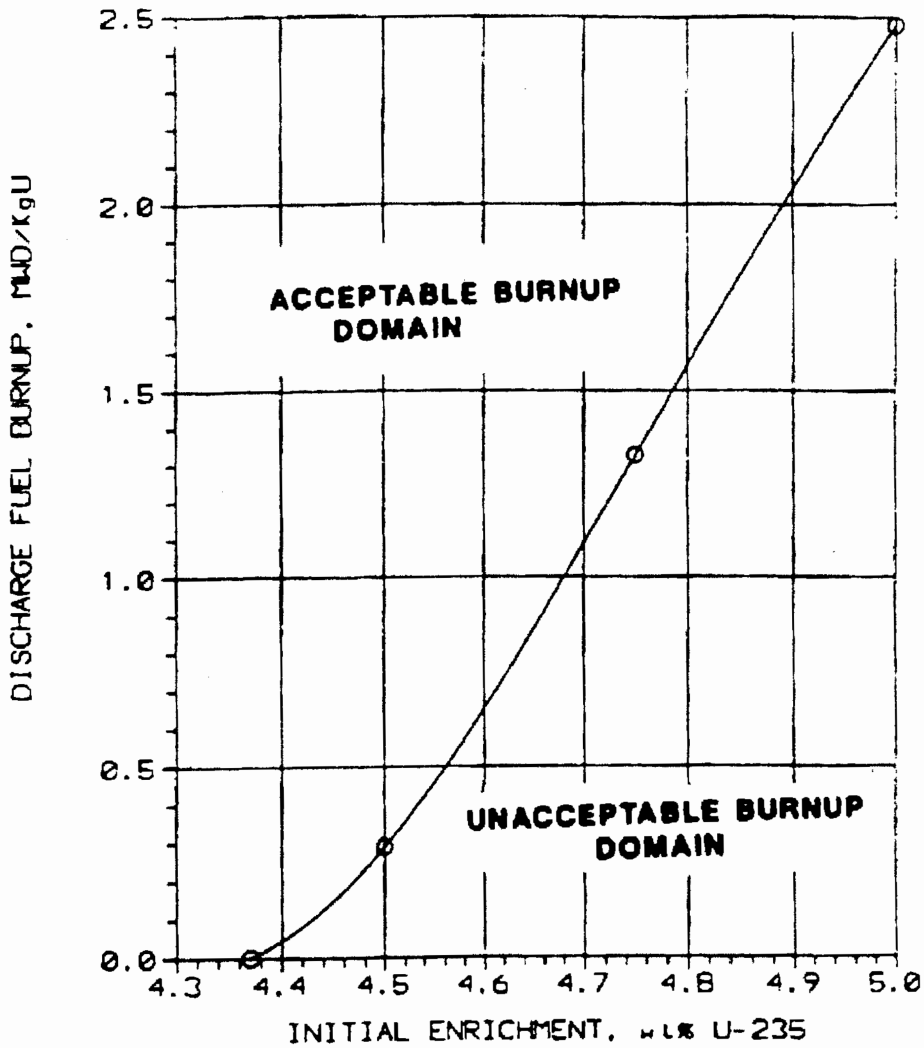

Figure 3.1.3-2
Minimum Burnup Requirements for Fuel in the Pool B Storage Racks

3/4.1.4 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

3.1.4.1 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.

3.1.4.2 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.

3.1.4.3 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.

3.1.4.4 Loads in excess of hook capacity shall not be lifted, except for load testing.

3.1.4.5 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.

3.1.4.6 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).

3.1.4.7 During transfer of the cask to and from the cask loading pit, the cask will be restricted to the transfer path shown in Figure 3.1.4-1. 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 3.1.4.1 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 3.1.4.2 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 3.1.4.3 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 3.1.4.4 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 3.1.4.5 confirms the load rating of the crane.

Specification 3.1.4.6 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 3.1.4.7 in combination with 3.1.4.3 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.

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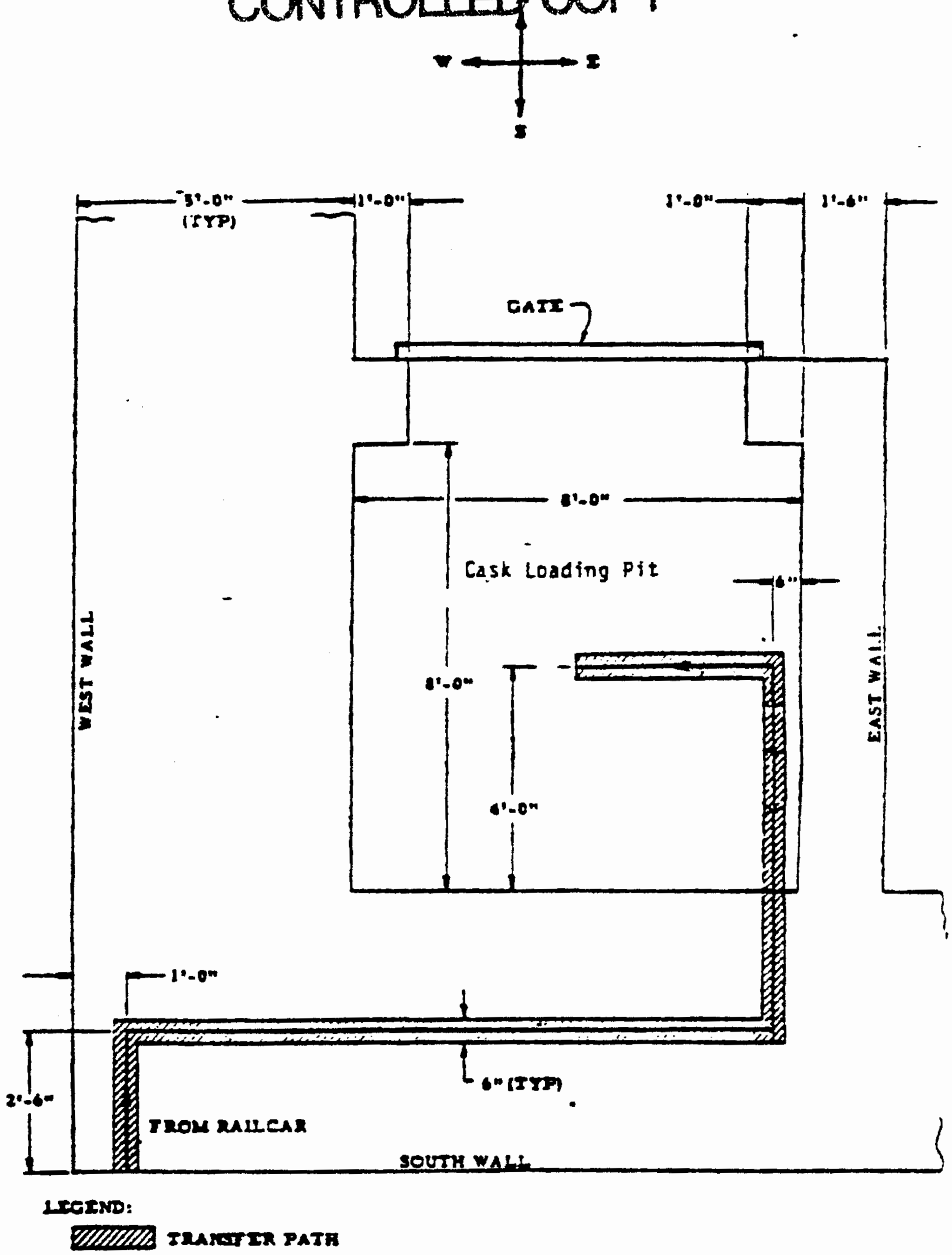

TRANSFER PATH TO AND FROM CASK LOADING PIT
(EL. 348"-0")
FIGURE 3.1.4-1

5.0 DESIGN FEATURES

5.1 <u>SITE</u>

<u>Applicability</u>

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

<u>Objective</u>

To define the above by location and distance description.

<u>Specification</u>

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.

5.2 SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for spent fuel assemblies.

Objective

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

Specification

5.2.1 SPENT FUEL STORAGE

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. 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 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 must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition.

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. 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.

c. 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.

REFERENCES

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

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,

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 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 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 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 due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

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. 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. 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.

6.9.2 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.2.1 The Annual Radiological Environmental Operating Report covering the 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.

6.9.3 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.3.1 The Annual Radioactive Effluent Release Report covering the 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 Process Control Program (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.

6.10 RECORD RETENTION

6.10.1 Records shall be retained as described by the Decommissioning Quality Assurance Program.

6.11 DELETED

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 DELETED

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).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 297 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

EXELON GENERATION COMPANY, LLC.

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated June 20, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17171A151), Exelon Generation Company, LLC (Exelon or the licensee) informed the U.S. Nuclear Regulatory Commission (NRC or the Commission) that the Three Mile Island Nuclear Station, Unit 1 (TMI-1), will permanently cease operations on or about September 30, 2019. Upon docketing of the certifications for permanent cessation of operations (paragraph 82(a)(1)(i) to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10 of the *Code of Federal Regulations* (10 CFR)), and permanent removal of fuel from the reactor vessel (10 CFR 50.82(a)(1)(ii)), pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for TMI-1 will no longer authorize operation of the reactor or the emplacement or retention of fuel into the reactor vessel.

By application dated July 25, 2018, as supplemented by letter dated March 6, 2019 (ADAMS Accession Nos. ML18206A545 and ML19065A217, respectively), Exelon requested changes to Renewed Facility Operating License No. DPR-50 (RFOL or the license) and the Technical Specifications (TSs) for TMI-1. Specifically, Exelon requested an amendment to revise the TMI-1 license and the associated TSs to Permanently Defueled Technical Specifications (PDTS), consistent with the permanent cessation of reactor operation and permanent defueling of the reactor. The proposed changes would also modify the current licensing basis mitigation strategies for flood mitigation and aircraft impact protection in the air intake tunnel.

The supplement dated March 6, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 20, 2018 (83 FR 58611).

2.0 REGULATORY EVALUATION

The NRC staff considered the following regulatory requirements and guidance in its review of the licensee's application.

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include TSs as part of their application. The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." Pursuant to 10 CFR 50.36, each operating license issued by the Commission includes TSs in the following categories: (1) safety limits (SLs), limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

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 focus on instrumentation to detect degradation of the reactor coolant system (RCS) pressure boundary and process variables; design features; operating restrictions; or structures, systems, or components (SSCs) that affect the integrity of fission product barriers during design-basis accidents (DBAs) or transients. They also focus on SSCs that operating experience or probabilistic risk assessment have shown to be significant to public health and safety. A general discussion of how these criteria were evaluated to ensure that the TS LCOs proposed for deletion are no longer required to be included in the TSs 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 is present in the reactor or RCS at TMI-1, 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 DBA 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. The scope of DBAs applicable to a permanently shutdown and defueled reactor is reduced from those postulated for an operating reactor, and most TSs satisfying Criterion 2 are no longer applicable. The one existing TS that defines the initial condition of the DBA associated with irradiated fuel movement is discussed in Section 3.0 of this safety evaluation.

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 functions or actuates 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 TSs 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. There are no transients that continue to apply to permanently shutdown and defueled reactors. The scope of applicable DBAs that continue to apply to TMI-1 is discussed in more detail in Section 3.0 of this safety evaluation.

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. There are no longer any DBAs at TMI-1 in the permanently shutdown and defueled condition that can result in a significant offsite radiological risk to public health and safety.

The NRC staff also considered the guidance in NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" (ADAMS Accession No. ML070570006).

The regulations in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distances," state, in part:

> (a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:
>
> (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.
>
> (2) A low population zone of such size that an individual located at any point on its outer boundary 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 total radiation dose to the whole body in excess of 25 rem [roentgen equivalent man] or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

---

[1] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis 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.

[2] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP [National Council on Radiation Protection and Measurements] recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

The regulations in 10 CFR 50.67(a)(2), "Accident source term," state, in part, that the NRC may issue a license 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 not receive a radiation dose in excess of 0.25 Sv (25 rem) 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) 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.

Appendix A, "General Design Criteria," to 10 CFR Part 50 (GDC), Criterion 19, "Control room," states as follows:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
>
> Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792), provides methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to

licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (ADAMS Accession No. ML003734190), provides review guidance to the staff for the review of AST amendment requests. Section 15.0.1 states that the NRC reviewer should evaluate the proposed changes against the guidance in RG 1.183. The dose acceptance criteria for the fuel handling accident (FHA) are a TEDE of 6.3 rem at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room for the duration of the accident.

NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347), discusses experiences with analyzing an accident involving a release from off-gas or waste systems. As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this type of accident, licensees have proposed acceptance criteria of 500 millirem (mrem) TEDE. The acceptance criteria for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

NUREG-0800, Section 11, Branch Technical Position 11-5, "Postulated Radioactive Release Due to a Waste Gas System Leak or Failure," Revision 4 (ADAMS Accession No. ML15027A302) provides guidance to the reviewer for assessing the analysis of an accidental release from the waste gas system.

License Amendment No. 235, dated September 19, 2001 (ADAMS Accession No. ML012480262), approves, in part, a full scope implementation of an AST for TMI-1 in accordance with 10 CFR 50.67 to perform the radiological consequences analyses of DBAs, as described in RG 1.183.

License Amendment No. 260 dated September 26, 2007 (ADAMS Accession No. ML072340348), revised the TMI-1 TSs to relocate the reactor building refueling area and spent fuel storage area radiation monitor operability requirements to the Updated Final Safety Analysis Report (UFSAR) and plant procedures based on the assertion that these radiation monitors do not meet the criteria for inclusion in the TSs as presented in 10 CFR 50.36(c)(2)(ii). To support the change, the licensee revised the TMI-1 FHA in the fuel handling building (FHB) described in UFSAR Section 14.2.2.1.b.1 without taking credit for FHB ventilation exhaust filtration. The updated analysis of the FHA in the FHB incorporates the AST methodology pursuant to 10 CFR 50.67, as described in RG 1.183.

3.0 TECHNICAL EVALUATION

3.1 Accident Analysis

Chapter 14 of the TMI-1 UFSAR describes the postulated DBA and transient scenarios applicable to TMI-1 during power operations. These scenarios demonstrate that the plant can be operated safely and that radiological consequences from postulated accidents do not exceed the regulatory guidelines of 10 CFR 50.67 or 10 CFR Part 100, as applicable. Two basic groups of events are pertinent to safety, which are abnormal operational transients and postulated DBAs; these two groups are 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.

The most severe postulated DBA involves damage to the nuclear reactor core and the release of large quantities of fission products. Many of these accident scenarios involve failures or malfunctions of systems, which could affect the fuel in the reactor vessel. With the termination of reactor operations and the permanent removal of fuel from the reactor vessel, such accidents are no longer possible. Therefore, the postulated accidents involving failure or malfunction of the reactor, reactor cooling system, steam system, or turbine generator, are no longer applicable. The licensee has stated, and the NRC staff agrees, that while spent fuel remains in the spent fuel pool (SFP), the accidents that remain applicable to TMI-1 in the permanently shutdown and defueled condition are the FHA within the SFP, the waste gas tank rupture (WGTR), and the fuel cask drop accident. The FHA within the reactor building is no longer an applicable concern. For completeness, the NRC staff also evaluated the applicability of other DBAs documented in the TMI-1 UFSAR to ensure that these accidents would not have consequences that could potentially exceed the 10 CFR 100.11 dose limits and RG 1.183 dose acceptance criteria.

Fuel Handing Accident Analysis

After the reactor has been completely defueled following permanent shutdown, an FHA in the reactor cavity is no longer a credible accident. The DBA FHA in the SFP pool is applicable when TMI-1 is in a permanently shutdown and defueled condition. The licensee's analysis applied the AST guidelines outlined in RG 1.183 and was performed to determine the dose to operators in the control room and the public at the EAB and low population zone as a function of time after shutdown. The analysis demonstrates that following 60 days of decay time after reactor shutdown, provided the SFP water level requirements of proposed TS LCO 3/4.1.1 are met, the dose consequences in the control room and at the EAB and LPZ are within allowable limits of 10 CFR 50.67, without relying on SSCs to remain functional for accident mitigation during and following the event.

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. 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. The

proposed TS LCO 3.1.1 will ensure the minimum water level in the SFP is established prior to fuel handling and maintained. The fuel release fractions from RG 1.183, Table 3, are conservatively doubled to bound fuel assemblies that potentially exceed the RG 1.183, footnote 11 value of 6.3 kilowatts/feet (kW/ft.) peak rod average power for burnups exceeding 54 gigawatt-days/metric ton of heavy metal. This is consistent with the licensing basis FHA analysis approved on September 26, 2007, in License Amendment No. 260 (ADAMS Accession No. ML072340348). Additionally, the Kr-85 and 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.

The activity released is assumed to reach the environment outside the building within 2 hours. The auxiliary and FHB ventilation system's exhaust discharges 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 FHB emergency ventilation system, (2) FHB 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 cubic feet per minute into and out of the control room is assumed to account for normal ingress and egress.

The analysis concludes that without crediting mitigation by any active SSC, the dose consequence of the post-permanent shutdown FHA at 60 days after reactor shutdown is within regulatory allowable limits for the DBA FHA.

In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff experience in performing similar reviews. The NRC staff concludes that the dose consequence from an FHA for the permanently defueled TMI-1 meets the applicable radiological dose criteria at the EAB, LPZ, and in the control room.

Waste Gas Tank Rupture

The waste gas disposal system collects, stores, monitors, samples, and releases radioactive gas, hydrogen, and oxygen from the primary coolant. 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 source term contained in the waste gas tanks is based on the activity of the primary coolant. The accident assumes a tank contains the gaseous activity evolved from degassing all of the reactor coolant following operation with 1 percent defective fuel. The reactor coolant passes through purification demineralizers that remove 99 percent of the iodine; however, no credit is assumed for this iodine removal. The coolant is then degassed an additional 99 percent according to the liquid/gas partitioning for iodine. The resulting waste gas inventory is 1 percent of the iodine and all of the noble gas activities associated with one reactor coolant volume.

Once the reactor is permanently shut down 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, which is expected to be less than the assumed WGTR analysis of record, and thus, bounded. 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. Additionally, the licensee explains that the new FHA analysis will bound the site doses for the WGTR because they will confirm the activity within any waste gas tank will be less than the FHA analysis of record prior to implementation of the PDTS.

The waste gas tank source term is assumed to be released to the auxiliary building and then to the environment as an instantaneous puff release. No radioactive decay is accounted for, and no removal mechanisms for noble gases are assumed. No credit is taken for any active safety system for the mitigation of the accident such as iodine removal by the auxiliary building ventilation charcoal filters.

The licensee explains that WGTR analysis of record, as described in the UFSAR, remains valid after permanent defueling. In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff experience in performing similar reviews. The NRC staff concludes that the dose consequence from a WGTR for the permanently defueled TMI-1 meets the applicable radiological dose criteria of 10 CFR Part 100 at the EAB and LPZ.

Fuel Cask Drop Accident

A fuel cask drop accident is defined as the dropping of a fuel cask through the maximum drop height during transfer operations of a fuel cask onto a rail car. A fuel cask drop into the SFP is prevented by the TS requirement that the key operated travel interlock system for automatically limiting the travel area of the FHB crane shall be imposed whenever loads more than 15 tons are lifted and transported.

The source term contained is based on the assumptions that the fuel cask and its entire contents of ten fuel assemblies are sufficiently damaged to allow the escape of all the noble gases and iodine in the gap activity of the primary coolant. The gap activity released from the fuel cask is based on a decay time of 120 days, which is the minimum time before final assemblies can be loaded into a cask, as required by the TSs. All 208 fuel pins in each of the 10 fuel assemblies are assumed ruptured.

The release of the noble gases and iodine is assumed to be directly to the atmosphere and to occur instantaneously. No credit is taken for any active safety system for the mitigation of the accident.

The licensee explains that fuel cask drop accident analysis of record, as described in the UFSAR, remains valid after permanent defueling. In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff experience in performing similar reviews. The NRC staff concludes that the dose consequence from a fuel cask drop accident for the permanently defueled TMI-1 meets the applicable radiological dose criteria of 10 CFR Part 100 at the EAB and LPZ.

3.1.1 NRC Staff Accident Analysis Conclusions

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The staff finds that the licensee's proposed

change to analysis methods and assumptions is consistent with the guidance contained in RG 1.183. The NRC staff compared the doses estimated by the licensee to the applicable criteria. The NRC staff finds with reasonable assurance that TMI-1, as modified by this proposed change, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters. The NRC staff concludes that the licensee has demonstrated that the dose consequences for postulated accidents at the permanently defueled TMI-1 would not have consequences that could potentially exceed the applicable 10 CFR 100.11 and 10 CFR 50.67 dose limits and RG 1.183 dose acceptance criteria. Therefore, the NRC staff finds the proposed changes to be acceptable from a dose consequence perspective.

3.2 Changes to Current Licensing Basis

3.2.1 Flood Mitigation Strategy

The licensee requested to modify its current licensing basis to no longer credit the flood barrier system to ensure SFP cooling is maintained in the event of a probable maximum flood (PMF). 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 licensee stated that the integrity of the FHB would not be 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 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 licensee stated that the SFP FLEX mitigation strategy will be maintained post-permanent shutdown and defueling.

If a PMF occurred and the dike and flood barrier system were not available, the normal means of SFP cooling would be lost but the integrity of the SFP would not be adversely affected. The licensee 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 adequate to reliably restore spent fuel cooling using the redundant components installed for post-Fukushima flood protection or equipment obtained from offsite if necessary. Given that the licensee will be maintaining the SFP FLEX mitigation strategy post-permanent shutdown and defueling and that the FHB would not be impacted by the PMF, the NRC staff finds that modifying the current licensing basis to no longer credit the flood barrier system to ensure SFP cooling is maintained in the event of a PMF is acceptable.

3.2.2 Protection from Aircraft Impact

The licensee requested to eliminate the licensing basis requirements in the current UFSAR 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.

The aircraft impact hazard and design protection are described in the TMI license basis in UFSAR Sections 1.2.7, 5.1.3, and 9.9.6. There are no TS LCOs 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.

As stated in UFSAR Sections 5.1.3 and 5.4.3.2.3, the FHB is designed for aircraft impact and would not have any adverse effect on SFP integrity or the ability to timely restore spent fuel level/cooling. Once the licensee 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). Currently the automatic detection and suppression in the AIT ensure control room habitability during an aircraft strike. After the reactor is permanently shut down and all fuel is in the SFP, the control room is not needed to conduct mitigative action to respond to such an event. 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 that 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 licensee 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 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 setup either of the two redundant sets of post-Fukushima spent fuel cooling components 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.

Based on the above, the NRC staff finds that modifying the current licensing basis to eliminate the automatic suppression of an explosion or fire in the AIT, and the general requirements for automatic ventilation shutdown or isolation of sump flow paths if combustible vapors are present, to be acceptable.

3.3 Changes to the License

3.3.1 License Condition 1.b

Currently, License Condition 1.b reads:

> 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;

The licensee proposes to delete License Condition 1.b because the decommissioning of TMI-1 does not depend on the conformity with Construction Permit No. CPPR-40. On April 19, 1974 (ADAMS Accession No. ML003762972), the Atomic Energy Commission issued Facility Operating License No. DPR-50 to the licensee. Construction Permit No. CPPR-40 was superseded by Facility Operating License No. DPR-50, which eventually became RFOL No. DPR-50, dated October 22, 2009 (ADAMS Accession No. ML092710401). Therefore, the NRC staff finds it acceptable to delete License Condition 1.b.

3.3.2 License Condition 1.c

Currently, License Condition 1.c reads:

> The facility will operate in conformity with the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

The licensee proposes License Condition 1.c to read:

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

The proposed change to the description "the facility will operate" to "the facility will be maintained" would provide a more accurate description of the requirements during the permanently shutdown and defueled condition. Once the licensee 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 proposed change is consistent with 10 CFR 50.82(a)(2) and is acceptable.

3.3.3 License Condition 1.d

Currently, License Condition 1.d reads:

> 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;

The licensee proposes License Condition 1.d to read:

> There is a reasonable assurance: (1) that the activities authorized by this renewed 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;

Once the licensee 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 word "operating" would provide accuracy in the 10 CFR Part 50 license description. Therefore, the NRC staff finds the proposed change acceptable.

3.3.4 License Condition 1.g

Currently, License Condition 1.g reads:

> 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;

The licensee proposes License Condition 1.g to read:

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

Once the licensee 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 word "operating" would provide accuracy in the 10 CFR Part 50 license description. Therefore, the NRC staff finds the proposed change acceptable.

3.3.5 <u>License Condition 1.h</u>

Currently, License Condition 1.h reads:

> 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;

The licensee proposes License Condition 1.h to read:

> After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of Renewed Facility 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;

Once the licensee 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 word "operating" would provide accuracy in the 10 CFR Part 50 license description. Therefore, the NRC staff finds the proposed change acceptable.

3.3.6 <u>License Condition 2</u>

Currently, License Condition 2 reads:

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

The licensee proposes License Condition 2 to read:

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

Once the licensee 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 word

"operating" would provide accuracy in the 10 CFR Part 50 license description. Therefore, the NRC staff finds the proposed change acceptable.

3.3.7 License Condition 2.a

Currently, License Condition 2.a reads:

> 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.

The licensee proposes License Condition 2.a to read:

> This renewed license applies to the Three Mile Island Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility), owned 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.

Once the licensee 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 words "and operated" would provide accuracy in the 10 CFR Part 50 license description. Therefore, the NRC staff finds the proposed change acceptable.

3.3.8 License Condition 2.b.(1)

Currently, License Condition 2.b.(1) reads:

> 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;

The licensee proposes License Condition 2.b.(1) to read:

> 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 the facility as required for fuel storage in accordance with the procedures and limitations set forth in this renewed license;

Once the licensee 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 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. The removal of the discussion of operating the facility would provide

accuracy in the 10 CFR Part 50 license description. Therefore, the NRC staff finds the proposed change acceptable.

3.3.9 License Condition 2.b.(2)

Currently, License Condition 2.b.(2) reads:

> 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;

The licensee proposes License Condition 2.b.(2) to read:

> Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to possess 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 in amounts as required;

The proposed change to this license condition removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel and limits the possession of SNM to SNM that was "used previously" as reactor fuel. The proposed change also 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. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that TMI-1 will no longer be authorized to operate and the continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. Once the licensee 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 a decommissioning plant. Therefore, the NRC staff finds the proposed change acceptable.

3.3.10 License Condition 2.b.(4)

Currently, the first sentence of License Condition 2.b.(4) reads:

> 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.

The licensee proposes the first sentence of License Condition 2.b.(4) to read:

> 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 that were produced by the operation of either unit.

This license condition is proposed for revision to allow possession, but not separation, of byproduct and SNM "that were" produced by the operation of the facility, as opposed to those materials "as may be" produced by the operation of the facility. Once the licensee 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 a decommissioning plant. Therefore, the NRC staff finds the proposed change acceptable.

3.3.11 <u>License Condition 2.c.(1)</u>

Currently, License Condition 2.c.(1) reads:

> <u>Maximum Power Level</u>
>
> Exelon Generation Company is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

The licensee proposes to delete License Condition 2.c.(1). Once the licensee 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); reference to operation of the facility would be inconsistent with 10 CFR 50.82(a)(2).

The NRC staff reviewed the proposed deletion of License Condition 2.c.(1) and determined that operation would not be authorized at TMI-1 at any power level once its 10 CFR 50.82(a)(1) certifications were docketed. Therefore, the NRC staff finds the proposed change acceptable.

3.3.12 <u>License Condition 2.c.(2)</u>

Currently, License Condition 2.c.(2) reads:

> <u>Technical Specifications</u>
>
> The Technical Specifications contained in Appendix A, as revised through Amendment No. 294, are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

The licensee proposes License Condition 2.c.(2) to read:

> <u>Technical Specifications</u>
>
> The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby incorporated in the license. Exelon Generation Company shall maintain the facility in accordance with the Permanently Defueled Technical Specifications (PDTS).

Once the licensee 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

discussion of operating license would provide accuracy in the 10 CFR Part 50 license description. This license condition is proposed for revision to account for the permanently defueled condition of the facility and to incorporate the PDTS. Therefore, the NRC staff finds the proposed change acceptable.

3.3.13 License Condition 2.c.(4)

Currently, License Condition 2.c.(4) reads:

> 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.

The licensee proposes to delete License Condition 2.c.(4). Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the fire protection program will be revised to take into account the facility conditions and activities during decommissioning. TMI-1 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 a fire protection program at an operating reactor in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire, will no longer be applicable at TMI-1. 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 shutdown and defueled facility is not necessary.

The NRC staff finds that License Condition 2.c.(4) for TMI-1 is based on maintaining fire protection programs that provide reasonable assurance of the ability to achieve and maintain safe shutdown in the event of a fire in accordance with 10 CFR 50.48. Achieving and maintaining safe shutdown in the event of a fire is no longer applicable to the decommissioned fire protection programs at TMI-1 once the facility is permanently shut down and the fuel has been permanently removed from the reactor. However, elements of the fire protection program (e.g., License Condition 2.C.(17), "Mitigating Strategy License Condition") continue during decommissioning to address fire events that could result in radiological hazards. The regulation in 10 CFR 50.48(f) requires TMI-1 to address the potential for fires that could result in a radiological hazard. The NRC staff concludes that the rule, which requires a fire protection program for licenses that have submitted the certifications under 10 CFR 50.82(a)(1), is sufficient to ensure that a program is maintained. Therefore, a license condition that also requires fire protection programs for the permanently shutdown and defueled unit is redundant. Based on the above, the NRC staff concludes that the licensee's request to delete License Condition 2.c.(4) is acceptable.

3.3.14 License Condition 2.c.(5)

Currently, License Condition 2.c.(5) reads:

> 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.

The licensee proposes to delete License Condition 2.c.(5). Once the licensee 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). A program to monitor secondary water chemistry designed to prevent steam generator tube degradation that may lead to a steam generator tube failure accident is no longer needed, as the postulated accident analyzed in UFSAR Chapter 14 is no longer credible. Therefore, the NRC staff finds the deletion of License Condition 2.c.(5) acceptable.

3.3.15 License Condition 2.c.(18)

Currently, License Condition 2.c.(18) reads:

> 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.

The licensee proposes to delete License Condition 2.c.(18). The license condition specifies specific time limits for initial completion of surveillances specified in License Amendment No. 264. The licensee stated that these assessments were completed in accordance with the schedule specified in the license condition. Since the requirements of this license condition have been completed, this license condition may be eliminated.

3.3.16 License Condition 2.c.(19)

Currently, License Condition 2.c.(19) reads:

> 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.

The licensee proposes to delete License Condition 2.c.(19). This license condition eliminates references to AmerGen Energy Company, LLC (AmerGen) and replaces them with references to Exelon Generation Company, LLC, to reflect the results of the license transfer. AmerGen transferred to Exelon ownership and control of AmerGen Three Mile Island NQF, LLC. 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 (ADAMS Accession No. ML082750072). 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 (ADAMS Accession No. ML090900463), Exelon reported to the NRC that the decommissioning trust agreements for TMI had been modified to reflect the change in ownership from AmerGen to Exelon. The requirements of this license condition have been completed; therefore, the NRC staff finds the deletion of License Condition 2.c.(19) acceptable.

3.3.17 License Condition 2.c.(20)

Currently, License Condition 2.c.(20) reads:

> 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.

The licensee proposes to delete License Condition 2.c.(20). This license condition was issued concurrent with the RFOL on October 22, 2009. This license condition is described in Section 1.7 of NUREG-1928, "Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1: Docket No. 50-289, Exelon Generation Company, LLC." 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 10 CFR 50.59 until the UFSAR update is completed. TMI-1 UFSAR, Revision 20, which included the supplement (Appendix A) for the license renewal application, was submitted to the NRC on December 10, 2010 (ADAMS Accession No. ML110420396). This action satisfied the requirements of TMI-1 License Condition 2.c.(20); therefore, the NRC staff finds the deletion of License Condition 2.c.(20) acceptable.

3.3.18 License Condition 2.c.(22)

The licensee proposes to add License Condition 2.c.(22) as follows:

> 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.

The licensee is proposing this license condition such that the initial activities to abandon unnecessary systems may be started expeditiously after the permanent removal of fuel from the reactor vessel. By applying this license condition, the licensee will be able to remove the TS requirements associated with those systems that perform mitigative actions assumed in the current licensing basis FHA by precluding the possibility of an 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 LCOs.

The deletion of the air filtration system LCOs in LCO 3.15 is 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 Attachment 1 to the application. 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, the licensee does not plan to handle or move irradiated fuel until it is relocated to the Independent Spent Fuel Storage Installation (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 expected to be required until after the completion of 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 was found to be erroneously loaded in a location not permitted by TS 5.4.2.g and TS 5.4.2.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, the licensee 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, the licensee 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 an FHA from occurring until after the 60-day decay period has elapsed.

For the reasons described above, the NRC staff finds this proposed license condition acceptable.

3.3.19 License Condition 2.d

Currently, License Condition 2.d reads:

> This license is effective as of the date of issuance and shall expire at midnight, April 19, 2034.

The licensee proposes License Condition 2.d to read:

> This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

The proposed change would modify this license condition to reflect the permanently shutdown and defueled condition of the facility. Once the licensee 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 proposed change would revise License Condition 2.d to conform with 10 CFR 50.51(b), "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.

The NRC staff reviewed the proposed change to License Condition 2.d. The current License Condition 2.d, which documents the date of the expiration of the license, is no longer necessary

for the permanently shutdown and defueled condition of the plant in the process of decommissioning. The revised License Condition 2.d is consistent with the provisions of 10 CFR 50.51(b) as applied to a facility that has permanently ceased operations. Therefore, the NRC staff finds the proposed change to License Condition 2.d acceptable.

3.4 Proposed TS Changes – TS Section 1, "Definitions"

3.4.1 Definitions Proposed for Deletion

The licensee proposes deleting the following definitions from TS 1.0 because they pertain to an operating reactor. Once the licensee 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):

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 refueling 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.

1.4 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 <u>INSTRUMENTATION SURVEILLANCE</u>

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.

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

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.

1.12 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.

1.17 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 any individual except when that individual is receiving an occupational dose.

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 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.

1.26 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).

In addition to the above changes, the licensee proposed to delete definition placeholders 1.9, 1.10, 1.11, and 1.14 currently in its TSs.

The NRC staff reviewed the TS definitions proposed for deletion and concludes that all the terms listed above are only meaningful for a reactor authorized to operate. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), and TMI-1 is permanently shut down and defueled, these definitions will no longer be necessary. Therefore, the NRC staff finds the deletion of the definitions from the TSs acceptable.

3.4.2 Definition Proposed for Addition

The licensee proposed addition of the following definition to clarify a term used in the remaining TS sections:

1.1 ACTIONS

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

The NRC staff reviewed the TS definition proposed for addition and concludes that the definition is meaningful for the defueled TSs and, therefore, is acceptable.

3.4.3 Definitions Proposed for Relocation

Currently, Definitions 1.15 and 1.16 read:

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.

The licensee proposed to relocate these definitions from the TSs to the ODCM.

The NRC staff reviewed the licensee's proposal to relocate the definitions for ODCM and PCP to the licensee-controlled document of the same name. The PDTS will continue to require that the ODCM contain the Radioactive Effluent Controls Program and the Radiological Environmental Monitoring Program and that the monitoring, sampling, analysis, and reporting of radiation and radionuclides in the gaseous effluent and the environment will be accomplished in accordance with the methodology and parameters in the ODCM. The acronyms ODCM and PCP will only be referenced in TS Section 6, "Administrative Controls," and will, therefore, be spelled out in that section. Therefore, the NRC staff finds that relocation of these definitions to the ODCM is acceptable.

3.4.4 Definition Proposed for Reformatting, Revision, and Relocation

Currently, Definition 1.25 reads:

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 percent of the surveillance interval. The 25 percent 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

The licensee proposed reformatting, revising, and relocating the definition of "frequency notation" to Section 3/4.0, "Limiting Conditions for Operation and Surveillance Requirement Applicability," as SR 4.0.3. Per the licensee, this will ensure the appropriate requirement for the 25 percent grace period is maintained. The portion of the definition with respect to frequency notation Table 1.2 is proposed for deletion due to elimination of most of the SRs.

The licensee proposes SR 4.0.3 to read:

> 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.

The NRC staff reviewed the licensee's proposal to reformat, revise, and relocate the definition of "Frequency Notation" to Section 3/4.0, "Limiting Conditions for Operation and Surveillance Requirement Applicability," as SR 4.0.3. The NRC staff determined that the proposed changes remove frequency notations that are no longer applicable to a facility in a permanently defueled condition, while ensuring the current TS allowance for a 25 percent grace period included in the definition of frequency notation is maintained. Therefore, the removal of the definition and addition of the new SR 4.0.3 is acceptable.

3.4.5 Definitions Proposed for Renumbering

The NRC staff reviewed the licensee's proposal to maintain the definitions for "Certified Fuel Handler"; "Non-certified Operator"; "Operable"; and "Station, Unit, Plant, and Facility," and renumber them to place them in alphabetic order with the remaining TS definitions. The NRC staff determined that these terms apply to the safe storage and handling of spent fuel in the SFP and are acceptable to be renumbered and retained in the defueled TSs.

3.5 Proposed TS Changes – TS Section 2, "Safety Limits and Limiting Safety System Settings"

The licensee proposed deletion of this section in its entirety. It includes the following TSs:

TS 2.1 – Reactor Core
TS 2.2 – Reactor System Pressure
TS 2.3 – Limiting Safety System Settings, Protective Instrumentation

The safety limits established in TS 2.1 and TS 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.

These TSs discussed above are being proposed for deletion in their entirety since the safety limits do not apply to a reactor that is in a permanently defueled condition. Once the licensee 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). These specifications do not apply to the safe storage and handling of spent fuel in the SFP. The NRC staff reviewed the specifications, as well as the associated bases, and finds that the safety limits and limiting safety system settings associated with the reactor core, reactor system pressure, and limiting safety system settings protective instrumentation are not applicable to a reactor in a permanently defueled condition. Therefore, the NRC staff finds the deletion of Section 2.0, "Safety Limits and Limiting Safety System Settings," in its entirety, acceptable.

3.6 Proposed TS Changes – TS Section 3, "Limiting Conditions for Operation"

3.6.1 General

This section contains LCOs that specify the lowest functional capability or performance levels of equipment required for safe operation of the facility.

The licensee proposed combining its LCOs (TS Section 3) with the corresponding SRs (TS Section 4) due to the reduced number of LCOs and SRs. The licensee proposed retitling the section header for the Section 3/4 combined TS Section as "3/4, 'Limiting Conditions for Operation and Surveillance Requirements.'"

The licensee proposed combining TS Sections 3.0 and 4.0 for TS LCO general action requirements and surveillance standards into a common specification. The licensee proposed retitling TS Section 3.0, "General Action Requirements," to "3/4.0, 'General Action Requirements and Surveillance Requirement Applicability.'"

The NRC staff reviewed the proposed changes and found that the changes are editorial in nature and do not impact the operation of the plant; therefore, the NRC staff finds the revisions acceptable.

3.6.2 TS Section 3/4.0, "General Action Requirements and SR Applicability"

The licensee proposed a revision to LCO 3.0.1 and addition of an LCO 3.0.2 in the PDTS as follows:

The current LCO 3.0.1 states the following:

> 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.

The licensee proposes revisions to this LCO in the proposed TS Section 3/4.0, "Limiting Conditions for Operation and Surveillance Requirement Applicability," to establish the applicability statement within each individual TS as the requirement for when the LCO is required to be met. This LCO is being revised to state the following:

3.0.1 LCOs shall be met during the specified conditions in the TS, except as provided in 3.0.2.

LCO 3.0.2 will state the following:

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.

The current LCO 3.0.1 pertains to an operating reactor. Since TMI-1 will be permanently shut down and defueled on or about September 30, 2019, this LCO no longer applies. The proposed LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated action shall be met. The NRC staff reviewed the proposed changes to LCO 3.0.1 and LCO 3.0.2. These LCOs are based on and are consistent with similar LCOs included in NUREG-1430, Volume 1, Revision 4, "Standard Technical Specifications Babcock and Wilcox Plants" (ADAMS Accession No. ML12100A177), as modified to reflect the permanently shutdown and defueled status of the facility. The NRC staff has previously determined that the LCOs included in NUREG-1430 are appropriate for plants similar in design to TMI-1. Therefore, the NRC staff finds the changes acceptable.

In addition to moving SR 4.0.1 and SR 4.0.2 to the proposed TS Section 3/4.0, the licensee also proposed revisions to SR 4.0.1 and SR 4.0.2. Also, as stated in Section 3.4.4 of this safety evaluation, Definition 1.25 was reformatted into SR 4.0.3. The NRC staff's review of that change is found in Section 3.4.4 of this safety evaluation.

The current SR 4.0.1 and SR 4.0.2 state the following:

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.

4.0.2 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. 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.

The proposed SR 4.0.1 and SR 4.0.2 read as follows:

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.

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 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.

The NRC staff reviewed the proposed changes to SR 4.0.1 and SR 4.0.2 and concluded that the changes are consistent with the transition to a permanently shutdown and defueled facility. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, the references to operability are no longer applicable. Further, the NRC staff agrees that the statements to be deleted are no longer necessary because the PDTS does not contain frequencies of the type described in the statements being deleted. In addition, the NRC staff finds the removal of performing a risk evaluation for a surveillance delayed greater than

24 hours acceptable due to the inherently lower risk associated with a permanently shutdown and defueled facility compared to an operating plant. Therefore, the proposed changes to delete these references would reflect the TMI-1 plant status and are appropriate, and the NRC staff finds the relocation of SR 4.0.1 and SR 4.0.2 and addition of SR 4.0.3 to PDTS Section 3/4.0 acceptable.

3.6.3 TS Section 3/4.1

Section 3.1 of the TMI-1 TSs, "Reactor Coolant System," contains the LCOs and actions to assure the operability of the 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 DBAs. This section contains the following LCOs:

- LCO 3.1.1 - Operational Components
- LCO 3.1.2 – Pressurizer Heatup and Cooldown Limitations
- LCO 3.1.3 – Minimum Conditions for Criticality
- LCO 3.1.4 – Reactor Coolant System Activity
- LCO 3.1.5 – Chemistry
- LCO 3.1.6 - Leakage
- LCO 3.1.7 – Moderator Temperature Coefficient of Reactivity
- LCO 3.1.8 – Single Loop Restrictions
- LCO 3.1.9 – Low Power Physics Testing Restrictions
- LCO 3.1.10 – Control Rod Operation – Previously deleted
- LCO 3.1.11 – Reactor Internals Vent Valves
- LCO 3.1.12 – Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)
- LCO 3.1.13 – Reactor Coolant System Vents

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.1 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.1 acceptable.

The licensee proposes to renumber and retitle TS 3.1 to TS 3/4.1, "Handling and Storage of Irradiated Fuel in the Spent Fuel Pool," and add three new specifications to address operability requirements for the SFP level, SFP boron concentration, and spent fuel assembly storage. The licensee is also proposing to renumber and rename TS 3.11 to TS 3/4.1.4. Specifically, the licensee proposes the following:

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.

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.

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.

The licensee proposed a new LCO 3/4.1.1 that will specify a minimum water level as expressed in SFP 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, which equates to a minimum elevation of water in the SFP of 342' 4". This TS provides the controls to ensure the assumptions of the accident analysis are met while fuel handling evolutions are in progress and satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). This TS includes SR 4.1.1.1, which will verify the SFP water level at 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.

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 FHA 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 TS 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.

Based on the above, the NRC staff finds that the SFP water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) and that the proposed addition of TS 3/4.1.1 is acceptable.

The licensee proposed the addition of TS 3/4.1.2 and TS 3/4.1.3 to ensure that the assumptions of storage limitations are maintained to prevent inadvertent criticality in the SFP. The technical content for these proposed TSs was taken directly from the current TS 5.4, "New and Spent Fuel Storage Facilities," including renumbering of Figures 5-4 and 5-5 to Figures 3.1.3-1 and 3.1.3-2. As the technical content remains unchanged, the NRC staff finds these additions acceptable, as they ensure the assumptions in the SFP criticality analysis continue to be met.

Based on the above, the NRC staff finds that the SFP water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) and that the proposed addition of TS 3/4.1.2 and TS 3/4.1.3 is acceptable.

TS LCO Section 3.11 is proposed to be renumbered and retitled as LCO 3/4.1.4, "Handling of Irradiated Fuel with the Fuel Handling Building Crane."

This title of the existing LCO is changed to reflect that the content of the TSs is the handling of fuel using the FHB crane versus the SFP fuel handling equipment. The content of the LCO will remain unchanged. These changes are deemed editorial in nature and are found acceptable by the NRC staff.

3.6.4 <u>Deletion of Current TS 3.3</u>

Section 3.3 of the TMI-1 TSs, "Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray System," 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 loss-of-coolant accident (LOCA) or isolation from the normal reactor heat sink. This section contains the following LCOs:

- LCO 3.3.1.1 – Injection Systems
- LCO 3.3.1.2 – Core Flooding System
- LCO 3.3.1.3 – Reactor Building Spray System and Reactor Building Emergency Cooling System
- LCO 3.3.1.4 – Cooling Water Systems
- LCO 3.3.1.5 – Engineered Safeguards Valves and Interlocks
- LCO 3.3.2 - 3.3.4 – Maintenance Requirements During Operation

The regulation in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specifies that applicable light-water nuclear power reactors must be provided with an emergency core cooling system designed to meet the cooling performance requirements following postulated LOCAs. Section 50.46(a)(1)(i) of 10 CFR states, in part, that, "[t]his 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."

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.3 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.3 acceptable.

3.6.5 Deletion of Current TS 3.4

Section 3.4 of the TMI-1 TSs, "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). This section contains the following LCOs:

- 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

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.4 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.4 acceptable.

3.6.6 Deletion of Current TS 3.5

Section 3.5 of the TMI-1 TSs, "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. This section contains the following LCOs:

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

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.5 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently

shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.5 acceptable.

3.6.7 Deletion of Current TS 3.6

Section 3.6 of the TMI-1 TSs, "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 provide shielding from the fission products that may be present in the containment atmosphere following accident conditions. This section contains the following LCOs:

- LCO 3.6.1 – Conditions that Require Containment Integrity (CI)
- LCO 3.6.2 – Conditions that Require CI with RCS open
- LCO 3.6.3 – Positive Reactivity Insertions
- LCO 3.6.4 – Reactor Building Internal Pressure Limits
- LCO 3.6.5 – Containment Isolation Valves (CIVs) Positions
- LCO 3.6.6 – Containment Isolation Valves (CIVs) Inoperable
- LCO 3.6.7 – Previously Deleted
- LCO 3.6.8 – 48" Reactor Building Purge Valve Inoperable
- LCO 3.6.9 – Previously Deleted
- LCO 3.6.10 – During Startup, Hot Standby and Power Operation Conditions
- LCO 3.6.11 – Reactor in Cold Shutdown or Refueling Shutdown conditions
- LCO 3.6.12 – Personnel or Emergency Air Locks

As previously discussed in Section 3.1 of this safety evaluation, the three DBAs that remain applicable to TMI-1 in the permanently shutdown and defueled condition are an FHA in the SFP, waste gas tank rupture, and a fuel cask drop accident. The analysis shows that radiological doses at the EAB, LPZ, and in the control room are within allowable limits of 10 CFR 50.67, without crediting secondary containment operability, standby gas treatment system, or control room high efficiency air filtration after a 60-day fuel decay period following permanent reactor shutdown for the FHA. The doses at the EAB and LPZ are within allowable limits of 10 CFR Part 100 for the waste gas tank rupture and a fuel cask drop accident.

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.6 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.6 acceptable.

3.6.8 Deletion of Current TS 3.7

Section 3.7 of the TMI-1 TSs, "Unit Electric Power System," contains LCOs related to the operability of alternating current (AC) and direct current (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

onsite standby power sources (emergency diesel generators), 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 emergency diesel generator fuel oil storage are included for each emergency diesel generator. Also included in this section are the requirements for DC power. This section specifies requirements to ensure that the DC electrical power subsystems are operable and contains the following LCOs:

- LCO 3.7.1 – Defines minimum electrical power systems requirements to place the reactor in a critical state
- LCO 3.7.2 – Defines the allowable outage times for electrical power systems while the reactor is critical

As previously discussed in Section 3.1 of this safety evaluation, the DBAs and transients analyzed in UFSAR Chapter 14 will no longer be applicable in the permanently shutdown and defueled condition, with the exception of the FHA in the SFP, waste gas tank rupture, and fuel cask drop accident. Exelon performed a calculation for an FHA in the SFP that shows that 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 permanent reactor shutdown and compliance with the SFP water level requirements in proposed TS 3/4.1.

Because the FHA analysis does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of irradiated fuel assemblies in the SFP for mitigation of a potential FHA. Therefore, 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 shutdown and defueled. As such, there are no DBAs that rely on AC and DC sources for mitigation.

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.7 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.7 acceptable.

3.6.9 <u>Deletion of Current TS 3.8</u>

Section 3.8 of the TMI-1 TSs, "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. This section contains the following LCOs:

- LCO 3.8.1 – Previously Deleted
- LCO 3.8.2 – Core Subcritical Neutron Monitors
- LCO 3.8.3 – Decay Heat Removal Pump and Cooler
- LCO 3.8.4 – Boron Concentration
- LCO 3.8.5 – Direct Communications Reactor Building to Control Room
- LCO 3.8.6 – Reactor Building Air-Lock Doors
- LCO 3.8.7 – Reactor Building Penetrations During Fuel Moves
- LCO 3.8.8 – Conditions to Stop Fuel Movement in the Core
- LCO 3.8.9 – Associated Radiation Monitors
- LCO 3.8.10 – No Irradiated Fuel Removed Until 72 Hours Subcritical
- LCO 3.8.11 – Maintain 23 Feet of Water Above RPV Flange

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.8 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.8 acceptable.

3.6.10 <u>Deletion of Current TS 3.9</u>

Section 3.9 of the TMI-1 TSs was previously removed. The remaining reference to TS 3.9 is being removed as part of the editorial cleanup of the TSs. The NRC staff finds the deletion of the reference to TS 3.9 acceptable.

3.6.11 <u>Deletion of Current TS 3.10</u>

Section 3.10 of the TMI-1 TSs, "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. This section contains the following LCOs:

- LCO 3.10.1.1 – The source leakage test performed pursuant to Specification 4.13
- LCO 3.10.1.2 – A complete inventory of licensed radioactive materials in possession shall be maintained current at all times

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.10 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.10 acceptable.

3.6.12 Deletion of Current TS 3.12

Section 3.12 of the TMI-1 TSs, "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. This section contains the following LCOs:

- LCO 3.12.1 – Reactor building crane operation during movement of fuel assemblies
- LCO 3.12.2 – During the period when the RPV head is removed
- LCO 3.12.3 – During the period when the RCS is pressurized

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.12 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.12 acceptable.

3.6.13 Deletion of Current TS 3.13

Section 3.13 of the TMI-1 TSs, "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 Tavg is greater than 200 °F. This section contains the following LCOs:

- LCO 3.13 – Secondary Coolant System Activity

As previously discussed in Section 3.1 of this safety evaluation, the DBAs and transients analyzed in UFSAR Chapter 14, including Main Steam Line Break, will no longer be applicable in the permanently shutdown and defueled condition, with the exception of the FHA in the SFP, waste gas tank rupture, and fuel cask drop accident.

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.13 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.13 acceptable.

3.6.14 Deletion of Current TS 3.14

Section 3.14 of the TMI-1 TSs, "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. 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 cubic feet per second, 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. This section contains the following LCOs:

- LCO 3.14.1 – Periodic inspection of the dikes around TMI
- LCO 3.14.2 – Flood condition for placing the unit in hot standby

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.14 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.14 acceptable.

3.6.15 Deletion of Current TS 3.15

Section 3.15 of the TMI-1 TSs, "Air Treatment Systems," contains requirements for the Emergency Control Room Air Treatment System, the Control Room Envelope (CRE) boundary, and the FHB ESF air treatment system. This section contains the following LCOs:

- 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 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 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 satisfies 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 greater than 200 °F, 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 DBAs that remain applicable relative to TMI TS in the permanently shutdown and defueled condition are an FHA in the SFP, waste gas tank rupture, and fuel cask drop accident. The post-permanent shutdown FHA analysis concluded that the dose consequences of an FHA 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, the licensee 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 2.c.(22). This will effectively prevent an 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 3.15.4 establishes the requirements for FHB ventilation during fuel movement and when the SRs associated with this LCO are met. 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, the licensee 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 an FHA from occurring until after the 60-day decay period has elapsed and allows this LCO to be eliminated during the decay period.

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.15 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.15 acceptable.

3.6.16 <u>Deletion of Current TS 3.16</u>

Section 3.16 of the TMI-1 TSs, "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 their support function. This section contains the following LCOs:

- LCO 3.16.1 – Safety-related snubber operability

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.16 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.16 acceptable.

3.6.17 <u>Deletion of Current TS 3.17</u>

Section 3.17 of the TMI-1 TSs, "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 LOCA. The limits also assure the maintenance of acceptable ambient environmental conditions for safety-related components located inside the containment. This section contains the following LCOs:

- LCO 3.17.1 – Primary containment average air temperature
- LCO 3.17.2 – Air temperature limits exceeded when critical
- LCO 3.17.3 – Primary containment average air temperature calculation

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.17 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.17 acceptable.

3.6.18 Deletion of Current TS 3.19

Section 3.19 of the TMI-1 TSs, "Containment Systems," establishes a requirement to verify containment structural integrity in accordance with the inservice tendon surveillance program for the reactor building prestressing system. This section contains the following LCOs:

- LCO 3.19.1 – Primary containment average air temperature
- LCO 3.19.2 – Previously deleted

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.19 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.19 acceptable.

3.6.19 Deletion of Current TS 3.20, TS 3.21, TS 3.22, and TS 3.23

Sections 3.20, 3.21, 3.22, and 3.23 of the TMI-1 TSs were previously removed by Amendment Nos. 139 and 197 (ADAMS Accession Nos. ML003765143 and ML003765832). The remaining references to TSs 3.20, 3.21, 3.22, and 3.23 are being removed as part of the editorial cleanup of the TSs. The NRC staff finds the deletion of the references to TSs 3.20, 3.21, 3.22, and 3.23 acceptable.

3.6.20 Deletion of Current TS 3.24

Section 3.24 of the TMI-1 TSs, "Reactor Water Level Indication," assures the operability of the reactor vessel water level indication instrumentation that may be useful in diagnosing situations that could represent or lead to inadequate core cooling. This section contains the following LCO:

- LCO 3.24 – Reactor vessel water level indication

The licensee proposes to delete the above-bulleted TSs. These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Section 3.24 will not be required, and these LCOs (and associated SRs in TS Section 4) will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 3.24 acceptable.

3.7 <u>Proposed TS Changes – TS Section 4, "Surveillance Standards"</u>

3.7.1 <u>Proposed Changes to TS 4.0</u>

As discussed in Section 3.6.1 of this safety evaluation, the licensee proposed combining TS Sections 3.0 and 4.0 for TS LCO General Action Requirements and Surveillance Standards into a common specification. As discussed in Section 3.6.2 of this safety evaluation, the licensee proposed revisions to SR 4.0.1 and SR 4.0.2, as well as the creation of SR 4.0.3, which was reformatted from Definition 1.25. The NRC staff's review of these changes can be found in Sections 3.6.1, 3.6.2, and 3.4.4 of this safety evaluation, respectively.

3.7.2 <u>Deletion of Current SR 4.1 through SR 4.23</u>

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

Since there are no safety limits that apply to TMI-1 with the reactor in a permanently shutdown and defueled condition, and since there are relatively few remaining LCOs, the number of corresponding SRs is proposed to be greatly reduced.

Specifically, the licensee proposes to delete (or delete reference to) the following SRs:

- SR 4.1 – Operational Safety Review
- SR 4.2 – Reactor Coolant System Inservice and Testing
- SR 4.3 – Deleted
- SR 4.4 – Reactor Building
- SR 4.5 – Emergency Loading Sequence and Power Transfer, Emergency Core Cooling System and Reactor Building Cooling System Periodic Testing
- SR 4.6 – Emergency Power Periodic Testing
- SR 4.7 – Reactor Control Rod System Tests
- SR 4.8 – Deleted
- SR 4.9 – Decay Heat Removal (DHR) Capability – Periodic Testing
- SR 4.10 – Reactivity Anomalies
- SR 4.11 – Reactor Coolant System Vents
- SR 4.12 – Air Treatment System
- SR 4.13 – Radioactive Materials Source Surveillance

- SR 4.14 – Deleted
- SR 4.15 – Main Steam System Inservice Inspection
- SR 4.16 – Reactor Internals Vent Valves Surveillance
- SR 4.17 – Shock Suppressors (Snubbers)
- SR 4.18 – Deleted
- SR 4.19 – Steam Generator (SG) Tube Integrity
- SR 4.20 – Reactor Building Air Temperature
- SR 4.21 – Radioactive Effluent Instrumentation
- SR 4.22 – Radioactive Effluents
- SR 4.23 – Radiological Environmental Monitoring

These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in SR 4.1 through SR 4.23 will not be required, and these SRs will not apply in a permanently shutdown and defueled condition. Therefore, the NRC staff finds the deletion of SRs 4.1 through 4.23 acceptable.

3.8 Proposed TS Changes – TS Section 5, "Design Features"

The existing TS Section 5, "Design Features," provides information and design requirements associated with plant systems. It includes the following TSs:

- TS 5.1 – Site
- TS 5.2 – Containment
- TS 5.3 – Reactor
- TS 5.4 – New and Spent Fuel Storage Facilities

3.8.1 Proposed Changes to TS 5.1.1

The current TS 5.1.1 states the following:

> 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 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).

The proposed TS 5.1.1 reads as follows:

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.

The licensee is proposing to remove specific details associated with the site boundary and instead reference the UFSAR. Other than Figures 5-2 and 5-3, which will be added to the UFSAR, the details being removed from TS 5.1.1 already exist in the UFSAR.

The regulation in 10 CFR 50.36(c)(4) states, "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of this section." The details being removed from TS 5.1.1, along with Figures 5-2 and 5-3, are related to exclusion areas, restricted areas, low population zone boundaries, and effluent release points. Since these details are not "materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety," they do not meet any of the criteria in 10 CFR 50.36(c)(4) and, therefore, the NRC staff finds the removal of the information and reference to the UFSAR to be acceptable.

3.8.2 Deletion of Current TS 5.2 and TS 5.3

The licensee is proposing to remove TS 5.2 and TS 5.3. TS 5.2 provides references to principal design parameters and applicable design codes for the reactor building and design standards for penetrations not serving accident-consequence limiting systems. TS 5.3 provides a description and requirements regarding the reactor core, fuel assemblies and control rod assemblies, and the RCS.

These TSs do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the protective functions addressed in TS Sections 5.2 and 5.3 will not be required and will not apply to the reactor in the permanently shutdown and defueled condition. Based on the above, the NRC staff finds the proposed deletion of current TS 5.2 and TS 5.3 acceptable.

3.8.3 Proposed Changes to TS 5.4

TS 5.4 is proposed to be renumbered as TS 5.2 due to the deletion of TS 5.2 and TS 5.3. The title of the specification is being revised to "Spent Fuel Storage Facilities," since the licensee will no longer receive or possess new fuel after it has permanently shut down and defueled. In the Applicability and Objective section of the TS, reference to "new" fuel assemblies is being removed as well.

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

TS 5.4.1.a is proposed to be renumbered as TS 5.2.1. The TS 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 the facility will no longer receive new fuel after it has permanently shut down and defueled. The notation of "ppmb" is revised to "ppm" [parts per million] since boron concentration is the specified parameter.

TS 5.4.1.b was previously deleted. The step is proposed to be removed for editorial TS cleanup.

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 renumbered and 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 renumbered to be TS 5.2.1.a.

TS 5.4.2.b is proposed to be deleted. The proposed TS LCO 3.1.2, "Spent Fuel Pool Boron Concentration," will specify the required boron concentrations in the SFP.

TS 5.4.2.c was previously deleted. The step is proposed to be removed for editorial TS cleanup.

TS 5.4.2.d is proposed to be editorially renumbered as TS 5.2.1.b.

TS 5.4.2.e is proposed to be editorially renumbered as TS 5.2.1.c.

TS 5.4.2.f was previously deleted. The step is proposed to be removed for editorial TS cleanup.

TS 5.4.2.g is proposed to be relocated. The TS 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." This is discussed in Section 3.6.3 of this safety evaluation.

TS 5.4.2.h is proposed to be relocated. The TS 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." This is discussed in Section 3.6.3 of this safety evaluation.

Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, TMI-1 will no longer receive new fuel assemblies and, therefore, does not require specifications related to new fuel storage. The NRC staff finds the proposed changes to TS 5.4 acceptable.

3.8.4 Deletion of Reference to TS 5.5

Section 5.5 of the TMI-1 TSs was previously removed. The remaining reference to TS 5.5 is being removed as part of the editorial cleanup of the TSs. The NRC staff finds the deletion of the reference to TS 5.5 acceptable.

3.9 Proposed TS Changes – TS Section 6, "Administrative Controls"

3.9.1 Deletion of Current TS 6.8.4.a.(2), TS 6.8.4.a.(3), TS 6.8.5, TS 6.9.2, TS 6.9.5, TS 6.9.6, TS 6.11, TS 6.13, TS 6.19, TS 6.20, and TS 6.21

The licensee proposes to delete the following TSs:

- TS 6.8.4.a.(2) – Land Use Census
- TS 6.8.4.a.(3) – Interlaboratory Comparison Program
- TS 6.8.5 – Reactor Building Leakage Rate Testing Program
- TS 6.9.2 – DELETED
- TS 6.9.5 – Core Operating Limits Report
- TS 6.9.6 – Steam Generator Tube Inspection Report
- TS 6.11 – Radiation Protection Program
- TS 6.13 – Process Control Plan
- TS 6.19 – Steam Generator (SG) Program
- TS 6.20 – Control Room Envelope Habitability Program
- TS 6.21 – Surveillance Frequency Control Program

These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the TMI-1 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2). Therefore, the specifications addressed in TS 6.8.4.a.(2), TS 6.8.4.a.(3), TS 6.8.5, TS 6.9.2, TS 6.9.5, TS 6.9.6, TS 6.11, TS 6.13, TS 6.19, TS 6.20, and TS 6.21 will not be required and will not apply in a permanently shutdown and defueled condition. Therefore, the NRC staff finds the deletion of these specifications acceptable.

3.9.2 Proposed Changes to TS 6.8.4.b

The licensee is proposing to change the term "Members of the Public" to "members of the public" to be consistent with the removal of Definition 1.22 as discussed in Section 3.4.1 of this safety evaluation. The NRC staff finds the proposed editorial change to TS 6.8.4.b acceptable.

3.9.3 Proposed Changes to TS 6.9.1

The licensee is proposing to remove previously "DELETED" portions of TS 6.9.1, "Routine Reports" (TS 6.9.1.A, TS 6.9.1.B.1, TS 6.9.1.B.3, TS 6.9.1.B.4, TS 6.9.1.B.5,

and TS 6.9.1.C) and renumber the remaining sections. There are no technical changes to TS 6.9.1; the changes are deletion and renumbering. Therefore, the NRC staff finds the proposed editorial changes to TS 6.9.1 acceptable.

3.9.4 Proposed Changes to TS 6.9.3 and TS 6.9.4

The licensee is proposing to remove "operation of the unit" and replace with "facility" for both current TS 6.9.3, "Annual Radiological Environmental Operating Report," and TS 6.9.4, "Annual Radioactive Effluent Release Report." Further, the licensee is proposing to renumber TS 6.9.3 and TS 6.9.4 to TS 6.9.2 and TS 6.9.3, respectively. This is due to the removal of TS 6.9.2.

Using the proposed term "facility" in place of "operation of the unit" is consistent and appropriate for a site undergoing decommissioning and will reflect the site is in a permanently defueled condition. The renumbering of TS 6.9.3 and TS 6.9.4 is not a technical change. Therefore, the NRC staff finds the changes to TS 6.9.3 and TS 6.9.4 acceptable.

3.9.5 Proposed Changes to TS 6.10

The licensee is proposing to delete the current requirements in TS 6.10, "Records Retention," for its records retention program and replace them with the following:

- 6.10.1 Records shall be retained as described by the Decommissioning Quality Assurance Program.

The current requirements for record retention are proposed to be deleted from PDTS on the basis that they are adequately addressed by the requirements in 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records." In addition, the requirements in 10 CFR Part 20, Subpart L, and 10 CFR 50.71, also require retention of certain records related to operation of the facility. Thus, the NRC staff finds removing the record retention requirements from the TSs and locating them in the Decommissioning Quality Assurance Program to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the official from the Commonwealth of Pennsylvania was notified of the proposed issuance of the amendment on July 29, 2019. The official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or an SR. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on November 20, 2018 (83 FR 58611). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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Date: August 29, 2019

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 – ISSUANCE OF AMENDMENT NO. 297 RE: DEFUELED TECHNICAL SPECIFICATIONS AND REVISED LICENSE CONDITIONS (EPID L-2018-LLA-0204) DATED AUGUST 29, 2019

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NAME	JPoole	LRonewicz	KHsueh	PSnyder
DATE	08/12/2019	08/05/2019	04/30/2019	05/07/2019
OFFICE	NRR/DSS/SRXB/BC**	NRR/DSS/SCPB/BC**	NRR/DSS/SNPB/BC*	NRR/DE/EEOB/BC*
NAME	JWhitman	SAnderson	RLukes	DWilliams
DATE	04/23/2019	05/20/2019	04/30/2019	08/08/2019
OFFICE	OGC*	NRR/DORL/LPL1/BC	NRR/DORL/LPL1/PM	
NAME	KGamin	JDanna	JPoole	
DATE	08/22/2019	08/29/2019	08/23/2019	

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