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

September 4, 2012

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

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION (TAC NO. ME8267)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 279 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated March 26, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12086A037), as supplemented by letter dated April 2, 2012 (ADAMS Accession No. ML12093A366).

The amendment revises TMI-1 Technical Specification (TS) Limiting Condition for Operation 3.1.1.2; TS Surveillance Requirement 4.19.2; TS 6.9.6, "Steam Generator Tube Inspection Report;" and TS 6.19, "Steam Generator (SG) Program;" changing certain inspection periods and making other administrative changes and clarifications. These changes are consistent with Technical Specification Task Force (TSTF) Traveler, TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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

Sincerely,

Peter Bamford

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

Docket No. 50-289

Enclosures:
1. Amendment No. 279 to DPR-50
2. Safety Evaluation

cc: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 279
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 March 26, 2012, as supplemented by additional letter dated April 2, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Renewed Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 279, are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena Khanna

Meena Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: September 4, 2012

ATTACHMENT TO LICENSE AMENDMENT NO.279

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove	Insert
Page 4	Page 4

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

Remove	Insert
3-1a	3-1a
3-1b	3-1b
4-77	4-77
6-19	6-19
6-20	6-20
6-26	6-26
6-27	6-27
6-28	6-28
--	6-28a

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No279 are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

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

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 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 275.

(4) Fire Protection

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

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

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

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;
b. Identification of the procedures used to measure the values of the critical parameters;
c. Identification of process sampling points;
d. Procedure for the recording and management of data;

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

3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator (SG) Tube Integrity

a. Whenever the reactor coolant average temperature is above 200°F, the following conditions are required:

(1.) SG tube integrity shall be maintained.

AND

(2.) All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program. (The Steam Generator Program is described in Section 6.19.)

ACTIONS:

--------------------NOTE--------------------

Entry into Sections 3.1.1.2.a.(3.) and (4.), below, is allowed for each SG tube.

(3.) If the requirements of Section 3.1.1.2.a.(2.) are not met for one or more tubes then perform the following:

With one or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program:

a. Verify within 7 days that tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, AND

b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to exceeding a reactor coolant average temperature of 200°F following the next refueling outage or SG tube inspection.

(4.) If Action 3., above, is not completed within the specified completion times, or SG tube integrity is not maintained, be in HOT SHUTDOWN within 6 hours and be in COLD SHUTDOWN within 36 hours.

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ± 1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

4.19 STEAM GENERATOR (SG) TUBE INTEGRITY

Applicability: Whenever the reactor coolant average temperature is above 200°F

Surveillance Requirements (SR):

Each steam generator shall be determined to have tube integrity by performance of the following:

4.19.1 Verify SG tube integrity in accordance with the Steam Generator Program.

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

BASES:

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by TS Section 3.4.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.19, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.19, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and

6.9.5 CORE OPERATING LIMITS REPORT

6.9.5.1 The core operating limits addressed by the individual Technical Specifications shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle.

6.9.5.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at TMI-1, specifically:

(1) BAW-10179 P-A, "Safety and Methodology for Acceptable Cycle Reload Analyses." The current revision level shall be specified in the COLR.

(2) TR-078-A, "TMI-1 Transient Analyses Using the RETRAN Computer Code", Revision 0. NRC SER dated 2/10/97.

(3) TR-087-A, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code", Revision 0. NRC SER dated 12/19/96.

(4) TR-091-A, "Steady State Reactor Physics Methodology for TMI-1", Revision 0. NRC SER dated 2/21/96.

(5) TR-092P-A, "TMI-1 Reload Design and Setpoint Methodology", Revision 0. NRC SER dated 4/22/97.

(6) BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", NRC SER dated February 4, 2000.

6.9.5.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient/accident analysis limits) of the safety analysis are met.

6.9.5.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.6 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the average reactor coolant temperature exceeds 200°F following completion of an inspection performed in accordance with Section 6.19, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

6.10 <u>RECORD RETENTION</u>

6.10.1 The following records shall be retained for at least five years:

a. Records of normal station operation including power levels and periods of operation at each power level.

b. Records of principal maintenance activities, including inspection, repairs, substitution, or replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of periodic checks, tests and calibrations.

e. Records of reactor physics tests and other special tests related to nuclear safety.

f. Changes to procedures required by Specification 6.8.1.

g. Deleted

h. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

i. Results of annual physical inventory verifying accountability of licensed sources on record.

j. Control Room Log Book.

k. Control Room Supervisor Log Book.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license or

   2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b.1 or 6.18.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71 (e).

6.19 STEAM GENERATOR (SG) PROGRAM

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational leakage performance criterion is specified in TS 3.1.6, "LEAKAGE."

c. Provisions for SG tube plugging criteria.

   1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
   b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
   c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
   d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary leakage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 279 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES

AND TUBE SAMPLE SELECTION

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated March 26, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12086A037), as supplemented by letter dated April 2, 2012 (ADAMS Accession No. ML12093A366), Exelon Generation Company (Exelon, or the licensee) requested changes to the Technical Specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The original submittal did not provide an application-specific no significant hazards consideration determination. The supplement, dated April 2, 2012, provided the missing information, which was then relied upon in the U.S. Nuclear Regulatory Commission (NRC) staff's proposed no significant hazards consideration determination, published in the *Federal Register* on May 15, 2012 (77 FR 28631).

The amendment revises TS Limiting Condition for Operation (LCO) 3.1.1.2; TS Surveillance Requirement (SR) 4.19.2; TS 6.9.6, "Steam Generator Tube Inspection Report;" and TS 6.19, "Steam Generator (SG) Program;" changing certain inspection periods and making other administrative changes and clarifications. These changes are consistent with Technical Specification Task Force (TSTF) Traveler, TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

TSTF Travelers, such as TSTF-510, evaluate changes to the Standard Technical Specifications (STSs). The STS applicable to the TMI-1 Nuclear Steam Supply System is NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants." The current STS provisions related to SG programs were established in May 2005 with the NRC staff's approval of TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC *Federal Register* Notice of Availability (70 FR 24126). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring that the integrity of the SG tubes is maintained. The performance-based provisions were supplemented by prescriptive provisions relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and

corrected on a timely basis. By letter dated September 27, 2007 (ADAMS Accession No. ML072600318), the NRC approved TSTF-449 for implementation in the TMI-1 TS.

After the issuance of TSTF-449, TSTF-510 was developed to reflect the industry's early implementation experience with respect to TSTF-449. TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability without changing the intent of the requirements. Further, according to the licensee's application, the proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections.

The following section details the regulatory requirements and guidance used by the NRC staff to evaluate the application.

2.0 REGULATORY EVALUATION

The SG tubes in Pressurized Water Reactors (PWRs) have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage during severe accidents.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) establish the requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have "an extremely low probability of abnormal leakage...and of gross rupture" (GDC 14), "shall be designed with sufficient margin to assure that the design conditions ... are not exceeded..." (GDC 15), "shall be designed with sufficient margin that when stressed ...(1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized" (GDC 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32). These GDC are referred to in TSTF-510.

The construction permit for TMI-1 was issued by the Atomic Energy Commission (AEC) on May 18, 1968, and an operating license was issued on April 19, 1974. The plant design approval for the construction phase was based on the proposed GDC published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A, 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "final GDC" or just "GDC"). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. In accordance with an NRC staff requirement memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which includes TMI-1.

The TMI-1 Updated Final Safety Analysis Report (UFSAR), Section 1.4, provides an evaluation of the design bases of TMI-1 against the draft GDC. The UFSAR evaluation of the draft GDC, specifically, Criterion 9, "Reactor Coolant Pressure Boundary;" Criterion 16, "Monitoring Reactor Coolant Pressure Boundary;" Criterion 33, "Reactor Coolant Pressure Boundary Capability;" Criterion 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention;" Criterion 35, "Reactor Coolant Pressure Boundary Brittle Fracture Prevention;" and Criterion 36, "Reactor Coolant Pressure Boundary Surveillance;" reflect design requirements similar to those specified in the final GDCs discussed in TSTF-510.

Paragraph 50.55a(c)(1) of 10 CFR specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code). Paragraph 50.55a(g)(4) of 10 CFR further requires, in part, that throughout the service life of a PWR facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements in Section XI, "Rules for Inservice Inspection [(ISI)] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

Section 50.36 of 10 CFR, "Technical specifications," establishes the requirements related to the content of the TS. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. As described in TSTF-510, LCOs and accompanying action statements and SRs in the STS relevant to SG tube integrity are in Specification 3.4.13, "Reactor Coolant System Operational Leakage," and Specification 3.4.17 (SR 3.4.17.2), "Steam Generator (SG) Tube Integrity." The SRs in the "Steam Generator (SG) Tube Integrity" specification reference the SG Program which is defined in the STS administrative controls. The TMI-1 TSs 3.1.6, "Reactor Coolant System Operational Leakage," and 3.1.1.2/4.19.2, "Steam Generator (SG) Tube Integrity," address requirements similar to those specified in the STS sections above.

Paragraph 50.36(c)(5) of 10 CFR defines administrative controls as "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee to operate the facility in a safe manner, including the SG Program, are listed in the administrative controls section of the TS. For TMI-1, the SG Program is defined in Specification 6.19, while the reporting requirements relating to implementation of the SG Program are in Specification 6.9.6.

Specification 6.19 requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Specification 6.19.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. SG tube integrity is maintained by meeting the performance criteria specified in TS 6.19.b for structural and leakage integrity, consistent with the plant design and licensing basis. The applicable tube repair criteria, specified in TS 6.19.c.1, are that tubes found during ISI to contain flaws with a depth equal to or exceeding 40 percent (%) of the nominal wall thickness shall be plugged. Specification 6.19.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that: (1) may be present along the length of a tube, from the tube-to-tubesheet weld at the

tube inlet to the tube-to-tubesheet weld at the tube outlet; and (2) may satisfy the applicable tube repair criteria.

3.0 TECHNICAL EVALUATION

Each proposed change to the TS is described individually below, followed by the NRC staff's assessment of the change.

3.1 Specification 6.19, "Steam Generator (SG) Program"

The last sentence in the introductory paragraph currently states, "In addition, the Steam Generator Program shall include the following provisions."

Proposed Change: The sentence is revised to say "In addition, the Steam Generator Program shall include the following:"

The licensee states that the subsequent paragraph starts with "Provisions for" and stating "provisions" in the introductory paragraph is duplicative.

Assessment: The NRC staff reviewed the licensee's proposed change to Specification 6.19 and has determined that the word "provisions" in the introductory paragraph is duplicative. The NRC staff has determined that the change is administrative in nature, changes no technical requirements, and therefore is acceptable.

3.2 Specification 6.19, Paragraph 6.19 b.1, "Structural integrity performance criterion"

The first sentence currently states:

> All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down, and all anticipated transients included in the design specification) and design basis accidents.

Proposed Change: Revise the sentence as follows:

> All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design-basis accidents.

According to the licensee, the basis for the change is that the sentence inappropriately includes anticipated transients in the description of normal operating conditions.

Assessment: The NRC staff has determined that the current wording is incorrect and that anticipated transients should be differentiated from normal operating conditions because they refer to separate and distinct parameters. Therefore, the NRC staff finds the change acceptable.

3.3. Paragraph 6.19.c, "Provisions for SG tube repair criteria," Paragraph 6.19.d, "Provisions for SG tube inspections," LCO 3.1.1.2, "Steam Generator (SG) Tube Integrity," and SR 4.19.2 for LCO 4.19, "Steam Generator Tube Integrity"

Proposed Change: Change all instances of the words "tube repair criteria" to "tube plugging criteria." According to the licensee, this change is intended to be consistent with the treatment of SG tube repair throughout Specification 6.19.

Assessment: The NRC staff finds that the proposed change provides a more accurate label of the criteria and, therefore, adds clarity to the specification. This is because generally, one of two actions must be taken when the criteria are exceeded. One action is to remove the tube from service by plugging the tube at both tube ends. The alternative action is to repair the tube, but only if such a repair is permitted in the TS by paragraph 6.19.c. TMI-1 does not have any approved alternate repair criteria, and thus plugging is the only available option if the criteria is exceeded. Therefore, the NRC staff finds the changes acceptable.

3.4 Paragraph 6.19.d, "Provisions for SG tube inspection"

Proposed Change: Change the term "assessment of degradation" to "degradation assessment" to be consistent with the terminology used in industry program documents.

Assessment: The NRC staff agrees that the terminology should be consistent. Further the proposed wording does not involve a technical change to the specification. Therefore, the NRC staff finds the change acceptable.

3.5 Paragraph 6.19.d.1

The paragraph currently states: "Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement."

Proposed change: The change would replace "SG replacement" with "SG installation." The basis for the change is that it will allow the SG Program to apply to both existing plants and new plants.

Assessment: The NRC staff agrees the SG Program can apply to both existing and new plants and the wording change allows for consistency between TMI-1 and other plants. Since this wording modification does not involve any technical or functional change for TMI-1, the NRC staff finds it acceptable.

3.6 Paragraph 6.19.d.2 (SGs with alloy 690 thermally treated tubes)

TSTF-510 is written to accommodate plants with several variations of SG tubing material. As described in the TMI-1 UFSAR, Revision 20, Chapter 4, Table 4.2-6, "Materials of Construction," as well as the licensee's letter regarding a previous TS amendment relating to the replacement SGs, dated April 2, 2009 (ADAMS Accession No. ML090920761), the TMI-1 SGs employ a thermally treated (TT) alloy 690 tubing design.

The inservice inspection status for the replacement TMI-1 SGs is described in the licensee's application dated March 26, 2012. The application states, "... the first SG inservice inspection following replacement was completed during the fall 2011 refueling outage (T1R19) in

accordance with Technical Specification 6.19.d.1. The first 144 effective full power month (EFPM) Inspection Period was entered following start-up from T1R19 as defined in the current and proposed Technical Specification 6.19.d.2."

Paragraph 6.19.d.2 currently states:

> Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

Proposed Change: Replace paragraph 6.19.d.2 with the following insert:

> After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.
>
> a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
>
> b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
>
> c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
>
> d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

Assessment:

Regarding paragraph 6.19.d.2, the licensee proposes, in part, to move the first two sentences, "Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs," of paragraph 6.19.d.2 to the inspection periods as specified in a, b, c and d of the revised paragraph, and make editorial changes to improve clarity. The NRC staff finds these changes to be of a clarifying nature, not changing the current intent of these two sentences. However, the license amendment request also includes three proposed changes to when inspections are performed as follows:

- The second inspection period would be revised from 108 to 120 EFPM.
- The third inspection period would be revised from 72 to 96 EFPM.
- The fourth and subsequent inspection periods would be revised from 60 to 72 EFPM.

The licensee characterizes these changes as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections. The NRC staff observes that depending on the actual plant inspection schedule, these changes could impact the number of inspections in a given period, as well as the sample size. However, inspection sample sizes will continue to be subject to paragraph 6.19.d.2, which states that in addition to meeting the requirements of paragraph 6.19.d.2, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure SG tube integrity is maintained until the next scheduled inspection. Therefore, the NRC staff concludes that with the proposed changes to the length of the second and subsequent inspection periods, compliance with the SG program requirements in Specification 6.19.d.2 will continue to ensure both adequate inspection scopes and tube integrity for the reasons addressed below.

For each inspection period, paragraph 6.19.d.2 currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the mid-point of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period, as compared to those in the second half, even when there are uniform intervals between each inspection. For example, a hypothetical plant in the second (120 EFPM) inspection period with a scheduled 36-month interval (two 18-month fuel cycles) between each inspection would currently be required to inspect 50 percent of the tubes by the refueling outage nearest the midpoint of the inspection period, which would be the third refueling outage in the period (after 54 EFPM), 6 months before the mid-point (assuming an inspection was performed at the very end of the 144 EFPM inspection period). However, since no inspection is scheduled for that outage (because inspections take place every other outage – once every 36 months), then the full 50 percent sample must be performed during the inspection scheduled for the second refueling outage in the period. Two inspections would be scheduled to occur in the second half of the inspection period, at 72 and 108 months into the inspection period. Thus, the current sampling requirement could be satisfied by performing a 25 percent sample during each of these inspections or other combinations of sampling (e.g., 10 percent during one and 40 percent in the other) totaling 50 percent. The NRC staff concludes that there is no basis for the minimum initial sample size to potentially have to vary so much from inspection to inspection. The

licensee proposes to revise this requirement such that the minimum sample size for a given inspection in a given inspection period is 100 percent divided by the number of scheduled inspections during that inspection period. For the above example, the proposed change would result in a uniform initial minimum sample size of 33.3 percent for each of the three scheduled inspections during the inspection period. The NRC staff concludes this proposed revision to be an improvement to the existing requirement, since it provides a more consistent minimum initial sampling requirement.

The proposed third and fourth sentences of paragraph 6.19.d.2 state, "If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period." This addresses the possibility that a degradation assessment in accordance with paragraph 6.19.d.2 will indicate that the tubing may be susceptible to a type of degradation at a location not previously inspected with a technique capable of detecting that type of degradation at that location. For example, new information from another similar plant becomes available, indicating the potential for circumferential cracking at a specific location on the tube. Previous degradation assessments had not identified the potential for this type of degradation at this location. Thus, previous inspections of this location had not been performed with a technique capable of detecting circumferential cracks. However, now that the potential for circumferential cracking has been identified at this location, paragraph 6.19.d.2 requires an inspection with a method capable of detection of a crack that may satisfy the applicable tube repair criteria.

Furthermore, suppose this inspection is performed for the first time during the third of four SG inspections scheduled for the 144 EFPM inspection period. In this case, the current paragraph 6.19.d.2 does not specifically identify whether 100 percent of the tubes at this location need to be inspected by the end of the 144 EFPM inspection period, or whether a prorated approach may be taken. The NRC staff addressed this question in Issue 1 of NRC Regulatory Information Summary (RIS) 2009-04, "Steam Generator Tube Inspection Requirements," dated April 3, 2009 (ADAMS Accession No. ML083470557), as follows:

> Issue 1: A licensee may identify a new potential degradation mechanism after the first inspection in a sequential period. If this occurs, what are the expectations concerning the scope of examinations for this new potential degradation mechanism for the remainder of the period (e.g., do 100 percent of the tubes have to be inspected by the end of the period or can the sample be prorated for the remaining part of the period)?
>
> [NRC Staff Position:] The TS contain requirements that are a mixture of prescriptive and performance-based elements. Paragraph "d" of these requirements indicates that the inspection scope, inspection methods, and inspection intervals shall be sufficient to ensure that SG tube integrity is maintained until the next SG inspection. Paragraph "d" is a performance-based element because it describes the goal of the inspections but does not specify

how to achieve the goal. However, paragraph "d.2" is a prescriptive element because it specifies that the licensee must inspect 100 percent of the tubes at specified periods.

If an assessment of degradation performed after the first inspection in a sequential period results in a licensee concluding that a new degradation mechanism (not anticipated during the prior inspections in that period) may potentially occur, the scope of inspections in the remaining portion of the period should be sufficient to ensure SG tube integrity for the period between inspections.

In addition, to satisfy the prescriptive requirements of paragraph "d.2" that the licensee must inspect 100 percent of the tubes within a specified period, a prorated sample for the remaining portion of the period is appropriate for this potentially new degradation mechanism. This prorated sample should be such that if the licensee had implemented it at the beginning of the period, the TS requirement for the 100 percent inspection in the entire period (for this degradation mechanism) would have been met. A prorated sample is appropriate because (1) the licensee would have performed the prior inspections in this sequential period consistently with the requirements, and (2) the scope of inspections must be sufficient to ensure that the licensee maintains SG tube integrity for the period between inspections.

The NRC staff finds that relocation of information in sentences 3 and 4, as described above, clarifies the existing requirement, such that it is consistent with the NRC staff's position from RIS 2009-04, and is therefore, acceptable.

The proposed fifth sentence in paragraph 6.19.d.2 states, "Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." Allowing extension of the inspection periods by up to an additional 3 EFPM potentially impacts the average tube inspection sample size to be implemented during a given inspection in that period. For example, if four SG inspections are scheduled to occur within the nominal 144 EFPM period, the minimum sample size for each of the four inspections could average as little as 25 percent of the tube population. If a fifth inspection can be included within the period by extending the period by 3 EFPM, then the minimum sample size for each of the five inspections could average as little as 20 percent of the tube population. Since the subsequent period begins at the end of the included SG inspection outage, the proposed change does not impact the required frequency of SG inspection.

Required tube inspection sample sizes are also subject to the performance-based requirement in paragraph 6.19.d.2, which states, in part, that in addition to meeting the requirements of paragraph 6.19.d.2, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next scheduled SG inspection." This requirement remains unchanged under the proposal. The NRC staff concludes the proposed fifth sentence, by allowing the potential for smaller sample sizes, involves only a relatively minor relaxation to the existing sampling requirements in paragraph 6.19.d.2. However, the performance-based requirements in 6.19.d.2 ensure that adequate inspection sampling will be performed to ensure tube integrity is maintained. Thus, the NRC staff concludes that the proposed change is acceptable.

Finally, the first sentence of the proposed revision to paragraph 6.19.d.2, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections)," replaces the last sentence of the current paragraph 6.19.d.2, "No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." Both versions establish the minimum allowable SG inspection frequency as at least every 72 EFPM or at least every third refueling outage (whichever results in more frequent inspections). This minimum inspection frequency in the proposed version is unchanged from the current requirement in the TMI-1 TSs. The NRC staff finds that the wording changes in the sentence are of an editorial and clarifying nature and are not material, such that the current intent of the requirement is unchanged. Thus, the NRC staff concludes the proposed change is acceptable.

3.7 Paragraph 6.19.d.3

The first sentence of paragraph 6.19.d.3 currently states:

> If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

Proposed Change: Revise this sentence as follows:

> If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

The proposed change is replacing the words "for each SG" with words "for each affected and potentially affected SG." The licensee states that the change should be made because the existing wording can be misinterpreted. The licensee further states that the intent is that those SGs that are affected and those SGs that are potentially affected must be inspected for the degradation mechanism that caused the crack indication. However, some licensees have questioned whether, perhaps, the current reference to "each SG" requires only the SGs that are affected to be inspected for the degradation mechanism. Therefore, according to the licensee, the proposed revision is intended to clarify the intent of the requirement – that it applies to both affected and potentially affected SGs.

Assessment: The proposed changes in paragraph 6.19.d.2 (Section 3.6) permits SG inspection intervals to extend over multiple fuel cycles for SGs with alloy 600 TT and 690 TT tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 6.19.d. However, stress-corrosion cracks may not become detectable by inspection until the crack depth approaches the tube repair limit. In addition, stress-corrosion cracks may exhibit high growth rates. For these reasons, once cracks have been found in any SG tube, paragraph 6.19.d.3 restricts the allowable interval to the next scheduled inspection to 24 EFPM or one refueling outage (whichever is less). The intent of this requirement is that it applies to the affected SG and to any other SG, which may be potentially affected by the degradation mechanism that caused the known crack(s). For example, a root

cause analysis in response to the initial finding of one or more cracks might reveal that the crack(s) are associated with a manufacturing anomaly which causes locally high residual stress, which in turn, caused the early initiation of cracks at the affected locations. If it can be established that the extent of condition of the manufacturing anomaly applies only to one SG and not the others, then the NRC staff agrees that only the affected SG needs to be inspected within 24 EFPM or one refueling cycle in accordance with paragraph 6.19.d.3. Conversely, if it cannot be established that the manufacturing anomaly applies to just one SG, then all potentially affected SGs would have to be inspected. The next scheduled inspections of the other SGs will continue to be subject to all other provisions of paragraph 6.19.d. The NRC staff finds the proposed change to paragraph 6.19.d.3 acceptable, because it clarifies the intent the paragraph.

3.8 Specification 6.9.6, "Steam Generator Tube Inspection Report"

This specification lists items a. through h. to be included in a report which shall be submitted within 180 days after the average reactor coolant temperature exceeds 200 [degrees Fahrenheit] °F following completion of an inspection performed in accordance with the Specification 6.19, "Steam Generator (SG) Program."

Proposed Change: Item b currently reads: "Active degradation mechanisms found." The proposed revision reads: "Degradation mechanisms found,".

Item e currently reads: "Number of tubes plugged during the inspection outage for each active degradation mechanism." The proposed revision reads: "Number of tubes plugged during the inspection outage for each degradation mechanism."

Item f currently reads: "Total number and percentage of tubes plugged to date." The proposed revision reads: "The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,".

The proposed change to item g replaces a comma at the end of the sentence with a period.

Item h currently reads: "The effective plugging percentage for all plugging [and tube repairs] in each SG," which, it is proposed, will be deleted because it is encompassed in the proposed item f.

Assessment: This proposal would delete the word "Active" in items b and e above. Thus, all degradation mechanisms found, whether deemed to be active or not, would now be reportable. The NRC staff finds the proposed change acceptable because it is more conservative. The change in item g and proposal to combine items f and h are editorial changes that do not materially change the reporting requirements. Thus, the NRC staff finds these changes acceptable.

3.9 Technical Conclusion

The NRC staff has reviewed the licensee's proposed changes and concludes that they are acceptable for the reasons described above. In the application dated March 26, 2012, the licensee included certain TS Bases changes for information only. TS Bases changes are controlled by the licensee, subject to the provisions of TS 6.18, "Technical Specification (TS) Bases Control Program." Hence, the NRC staff review of this application does not involve

approval, or disapproval, of the submitted TS Bases changes. Additionally, any TS Bases information included on the TS pages issued with this amendment, due to the format of the custom TMI-1 TSs, remains under the control of the TMI-1 TS Bases Control Program.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and an inspection or surveillance requirement. 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 (77 FR 28631). 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.

Principal Contributors: R. Grover
K. Karwoski
P. Bamford

Date: September 4, 2012

September 4, 2012

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

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION (TAC NO. ME8267)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 279 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated March 26, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12086A037), as supplemented by letter dated April 2, 2012 (ADAMS Accession No. ML12093A366).

The amendment revises TMI-1 Technical Specification (TS) Limiting Condition for Operation 3.1.1.2; TS Surveillance Requirement 4.19.2; TS 6.9.6, "Steam Generator Tube Inspection Report;" and TS 6.19, "Steam Generator (SG) Program;" changing certain inspection periods and making other administrative changes and clarifications. These changes are consistent with Technical Specification Task Force (TSTF) Traveler, TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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

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

Docket No. 50-289

Enclosures:
1. Amendment No. 279 to DPR-50
2. Safety Evaluation
cc: Distribution via Listserv

DISTRIBUTION
PUBLIC
RidsAcrsAcnw_MailCTR Resource
RidsNrrDorlDpr Resource
RidsRgn1MailCenter Resource
RidsNrrLAABaxter Resource
RidsNrrDirsItsb Resource
RidsNrrDorlLpl1-2 Resource
RidsNrrPMThreeMileIsland Resource
LPLI-2 R/F
RGrover, NRR
RidsOgcRp Resource
KKarwoski, NRR

ADAMS Accession No.: ML12209A392 * via memo ** via email

OFFICE	LPL1-2/PM	LPLI-2/LA	ITSB/BC	OGC	LPL1-2/BC
NAME	PBamford	ABaxter **	RElliott *	JLindell, NLO (w/ comments)	MKhanna
DATE	7/30/12	08/13/12	08/10/12	8/24/12	9/4/12

Official Record Copy