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

September 14, 2011

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: MAXIMUM ALLOWABLE POWER WITH INOPERABLE MAIN STEAM SAFETY VALVES (TAC NO. ME4808)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 277 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 September 24, 2010,[1] as supplemented by additional letters.[2]

The amendment revises Technical Specification 3.4.1.2.3, to allow up to two Main Steam Safety Valves (MSSVs) per steam generator to be inoperable with no required reduction in power level. It also revises the required maximum overpower trip setpoints for any additional inoperable MSSVs consistent with the plant transient analysis. The change requires that with less than four MSSVs associated with either steam generator operable, the plant would be required to be brought to the hot shutdown condition.

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. 277 to DPR-50
2. Safety Evaluation

cc: Distribution via Listserv

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1. Agencywide Documents Access and Management System (ADAMS) Accession No. ML102780570.
2. Letters dated March 18, 2011 (ADAMS Accession No. ML110770296), April 21, 2011 (ADAMS Accession No. ML111120003), and May 27, 2011 (ADAMS Accession No. ML111920354).

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. 277
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 September 24, 2010,[1] as supplemented by additional letters,[2] 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.

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1. Agencywide Documents Access and Management System (ADAMS) Accession No. ML102780570.
2. Letters dated March 18, 2011 (ADAMS Accession No. ML110770296), April 21, 2011 (ADAMS Accession No. ML111120003), and May 27, 2011 (ADAMS Accession No. ML111920354).

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. 277, 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

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

Attachment: Changes to the License and Technical Specifications

Date of Issuance: September 14, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 277

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 page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3-26a	3-26a

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 277 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.

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY (Continued)

3.4.1.2.3 Except as provided in Specification 3.4.1.2.2 above, when the Reactor is above HOT SHUTDOWN, seven (7) MSSVs per OTSG shall be OPERABLE. If either OTSG has less than seven (7) MSSVs that are OPERABLE, then reduce the power and reset the maximum overpower trip setpoint as follows:

Minimum Number of MSSVs Operable on Each OTSG	Maximum Overpower Trip Setpoint (% of Rated Power)
7	see Table 2.3-1
6	85.1
5	70.1
4	55.1

With less than four (4) MSSVs OPERABLE per OTSG, restore to a condition with at least four (4) MSSVs on each OTSG to OPERABLE status within 4 hours or be in HOT SHUTDOWN within the next 6 hours.

3.4.2 RCS temperature less than or equal to 250 degrees F.

3.4.2.1 At least two of the following means for maintaining DHR capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. DHR String (Loop "A").

b. DHR String (Loop "B").

c. RCS Loop "A" and its associated OTSG with an EFW Pump and a flowpath.

d. RCS Loop "B" and its associated OTSG with an EFW Pump and a flowpath.

With less than the above required means for maintaining DHR capability OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

3.4.2.2 Operation of the means for DHR may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for DHR required to be OPERABLE per Specification 3.4.2.1 may be reduced to one provided that the Reactor is in a REFUELING SHUTDOWN condition with the Fuel Transfer Canal water level greater than or equal to 23 feet above the Reactor Vessel flange.

3.4.2.4 Specification 3.4.2.1 does not apply when either of the following conditions exist:

a. Decay heat generation is less than 188 KW with the RCS full.

b. Decay heat generation is less than 100 KW with the RCS drained down for maintenance.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 277 TO FACILITY OPERATING LICENSE NO. DPR-50

MAXIMUM ALLOWABLE POWER WITH INOPERABLE MAIN STEAM SAFETY VALVES

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated September 24, 2010,[1] as supplemented by additional letters,[2] Exelon Generation Company, LLC (Exelon, or the licensee), requested changes to the Technical Specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards determination as published in the *Federal Register* on November 30, 2010 (75 FR 74096).

The amendment revises TS 3.4.1.2.3, to allow up to two Main Steam Safety Valves (MSSVs) per steam generator to be inoperable with no required reduction in power level. It also revises the required maximum overpower trip setpoints for any additional inoperable MSSVs consistent with the plant transient analysis. The change requires that with less than four MSSVs associated with either steam generator operable, the plant would be required to be brought to the hot shutdown condition.

2.0 REGULATORY EVALUATION

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 General Design Criteria (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 to Title 10 of the *Code of Federal Regulations* (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

1. Agencywide Documents Access and Management System (ADAMS) Accession No. ML102780570.
2. Letters dated March 18, 2011 (ADAMS Accession No. ML110770296), April 21, 2011 (ADAMS Accession No. ML111120003), and May 27, 2011 (ADAMS Accession No. ML111920354).

Enclosure

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 following regulatory requirements and guidance were used to evaluate the application:

10 CFR Part 50, Appendix A, GDC-10 "Reactor Core Design," states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Draft GDC, Criterion 6, applies to TMI-1 and contains similar statements.

10 CFR Part 50, Appendix A, GDC-15, "Reactor Coolant System Design," states that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. GDC-15 does not have a direct companion in the Draft GDC. However, draft GDC, Criterion 9, contains similar statements, and the TMI-1 UFSAR states that TMI-1 meets Draft GDC-9, in part, through compliance with American Society of Mechanical Engineers (ASME) and United States of America Standard (USAS) codes.

NUREG-0800, Standard Review Plan (SRP), Section 10.3, "Main Steam Supply System," provides review guidance, functional capabilities, and acceptance criteria for design of the main steam system and components. Section 5.2.2, "Overpressure Protection," provides guidance on the prevention of overpressurization of the reactor coolant pressure boundary.

ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Article NC-7000, "Overpressure Protection," provides requirements for the lift setpoints of main steam safety valves.

3.0 TECHNICAL EVALUATION

3.1 System Description

In the UFSAR, Chapter 10, the TMI-1 main steam system is described. Its purpose is to remove heat energy from the reactor coolant system (RCS) by transporting steam from the two once-through steam generators (OTSGs) via a total of four main steam lines, to the high pressure turbine during power operation. The OTSGs comply with the ASME Code, Section III, and have a secondary side design pressure of 1150 pounds per square inch gauge (psig). The balance of piping in the main steam system is designed in accordance with the USAS Code B31.1.0 (B31.1), and has a design pressure of 1050 psig. The TMI-1 UFSAR, Section 10.3.1.1 states that the MSSVs are designed in accordance with ASME Code Section III, Class A requirements.

The MSSVs are located upstream of the main steam isolation valves and individually have no isolation valves; therefore, the MSSVs are always available. There are two main steam lines from each of the two OTSGs. Each OTSG has nine MSSVs; five on one line and four on the other, for a plant total of eighteen MSSVs. Sixteen MSSVs are 6" x 10" and two are 3" x 6". The lift setpoints are staggered. The two smallest MSSVs start relieving at 1040 psig, then six relieve at 1050 psig, four more at 1060 psig, four more at 1080 psig, and the last two relieve at 1092.5 psig. Together, the MSSVs function to provide independent high pressure relief by venting sufficient steam to protect the RCS, OTSGs, and the main steam system from overpressurization.

3.2 Proposed Facility Operating License and Technical Specifications Changes

The licensee proposes to change TS 3.4 to allow a certain number of MSSVs to be inoperable according to the reactor power level. If more than two MSSVs are inoperable on one OTSG, then reactor power must be reduced accordingly.

3.2.1 Current TS

Currently the licensee's TS 3.4, "Decay Heat Removal (DHR) Capability," states under provision 3.4.1.2.3

> Except as provided in Specification 3.4.1.2.2 above, when the Reactor is above HOT SHUTDOWN, all eighteen (18) MSSVs shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of MSSVs Disabled on Any OTSG	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

> With more than three (3) MSSVs inoperable, restore at least fifteen (15) MSSVs to OPERABLE status within 4 hours or be in HOT SHUTDOWN within the next 6 hours.

3.2.2 Proposed TS

The licensee proposes changing the maximum power levels associated with the number of MSSVs allowed out of service on each steam generator, and total number of MSSVs allowed to be inoperable. The licensee proposes changing TS 3.4.1.2.3 to read as follows:

> Except as provided in Specification 3.4.1.2.2 above, when the Reactor is above HOT SHUTDOWN, seven (7) MSSVs per OTSG shall be OPERABLE. If either OTSG has less than seven (7) MSSVs that are OPERABLE, then reduce power and reset the maximum overpower trip setpoint as follows:

Minimum Number of MSSVs Operable on Each OTSG	Maximum Overpower Trip Setpoint (% of Rated Power)
7	See Table 2.3-1
6	85.1
5	70.1
4	55.1

With less than four (4) MSSVs OPERABLE per OTSG, restore to a condition with at least four (4) MSSVs on each OTSG to OPERABLE status within 4 hours or be in HOT SHUTDOWN within the next 6 hours.

3.3 NRC Staff Evaluation

3.3.1 Evaluation of Inoperable MSSVs on System Pressures

In accordance with NUREG-0800, Sections 10.3 and 5.2.2, the staff reviewed the submittal to ensure that the proper design transient was used in establishing the flow capacity and setpoints of the MSSVs. The safety functions of the main steam system are to provide a heat sink during transients and accidents in order to limit the RCS pressure, and to provide overpressure protection for the OTSGs and main steam piping. The pressurizer, its safety valves, MSSVs, reactor protection system, and other safeguards systems work together to protect the primary and secondary systems against overpressure in the event of a loss of heat sink.

In the submittal dated September 24, 2010, and supplemented by letter dated May 27, 2011, the licensee provided AREVA Calculation No. 86-9054640, which documented the effects on the pressures in the RCS and main steam system by varying the number of MSSVs inoperable. The AREVA calculation used a reactor power of 2772 Megawatts thermal (MWt), based upon a potential future increase in reactor power; however, the higher power evaluation bounds the current licensed maximum reactor power of 2568 MWt. The licensee used the computer code RELAP5/MOD2-B&W (R5/M2-B&W) to evaluate the system's response to a turbine trip with various numbers of MSSVs inoperable.

R5/M2-B&W was approved by the NRC to perform non-loss-of-coolant-accident (non-LOCA) safety analyses via the approval of topical report BAW-10164PA-06, "RELAP5/MOD2-B&W for Safety Analysis of [Babcock and Wilcox] B&W Designed Pressurized Water Reactors" (ADAMS Accession No. ML071620460). The basis for the NRC staff's approval included a comparison of R5/M2-B&W results to OTSG test facility data, to benchmarking data gained from actual reactor operation, and to results of similar analyses performed using other computer codes. The test data included steam generator heat transfer data under various upset conditions; while the operating data included the loss-of-feedwater event from TMI-2, which started initially with a steam generator pressurization and reactor coolant heatup similar to that which would result from the analyzed pressurization transients. The agreement in all tests was observed to be generally good, and was found by the staff to be acceptable. For this license amendment request (LAR) review, the staff concludes that the use of the topical report is applicable to the analyzed pressurization events because this report modeled events at plants similar in design to TMI-1 and because the topical report modeled events with similar initial behavior as the events analyzed for this LAR.

Design and testing requirements for ASME Section III components are specified in Article NC-7000, "Overpressure Protection." The code requires that total relieving capacity shall be sufficient to prevent a rise in pressure of no more than 10% above the design pressure of any component within the systems' pressure boundary. The licensee established an acceptance criteria based upon the maximum pressure design rating for the OTSGs and the main steam lines. For the OTSGs, the licensee used ASME Section III (2001 edition) to establish a design maximum of 110% of the 1150 psig, (1279.7 pounds per square inch absolute (psia)). For the main steam lines, the analysis acceptance criteria was set at 110% of 1050 psig (1169.8 psia). This bounds the allowable USAS B31.1 (1967 edition) piping code value of 120% of 1050 psig. Regarding RCS pressure, the TMI-1 UFSAR Section 14.1.2.8, states that the loss of load event must limit RCS pressure to less than 110% of design pressure (design pressure equals 2500 psig). The criteria set in the analysis (2764.7 psia) bounds this value. As a further measure of conservatism, the analysis set an acceptance criteria of 2707.69 psia which is the calculated startup accident peak pressure. Therefore, the analysis acceptance criteria was set at or below the UFSAR and ASME/USAS code limits for each of the systems and components being protected.

The ASME Code, Section III, Paragraph NC-7410, Set Pressure Limitations for Upset Conditions, states, "the stamped set pressure of at least one of the pressure relief devices [MSSVs] connected to the system shall not be greater than the design pressure of any component within the pressure retaining boundary of the protected system." Since the licensee requested to remove a number of MSSVs during power operation, it was possible to remove the MSSVs with lift setpoints below the main steam line design rating of 1050 psig. Based on this discrepancy, the staff asked the licensee to explain why it was acceptable to potentially remove MSSVs from service with lift setpoints below 1050 psig.

In a letter dated April 21, 2011, the licensee responded to the staff's request for additional information (RAI) regarding this question, by explaining that the requirements of ASME Code Section III apply only to the OTSGs, whose design pressure is 1150 psig. Since all MSSV lift setpoints are lower than 1150 psig, the licensee maintains the requirements of ASME Code Section III for the OTSGs are satisfied. Also, the licensee explained that the main steam piping is designed under USAS B31.1 which does not have the same code requirement for safety valve setpoints as ASME Section III. Under USAS B31.1, overpressure protection is set at 20% above design pressure for events occurring less than 1% of the operating period. The licensee conservatively used 10% above 1050 psig (1169.7 psia or 1155 psig) as the acceptance criteria for the main steam piping to ensure that USAS B31.1 code requirement is not exceeded. The NRC staff compared the code requirements with the licensee's position and licensing basis. The staff agrees that the proposed TS change will continue to satisfy the code requirements that are applicable to the OTSGs and the main steam piping.

The licensee's evaluation shows with a certain number of MSSVs inoperable at selected power conditions, the plant will not exceed the 110% of the design maximum pressure in the RCS, OTSGs, or main steam lines. The licensee's calculation shows that no more than two MSSVs per OTSG are allowed to be inoperable at full power conditions. The licensee's analyses further determined that power operation may continue, but at a specified reduced power with up to five MSSVs inoperable on one OTSG. The licensee determined the most limiting event for maximizing secondary system pressure was a turbine trip without actuation of a runback signal, or activation of the anticipatory reactor trip system (ARTS). In such an event, once the turbine stop valves close, pressure in the main steam lines rapidly increases, resulting in lifting of the MSSVs. In this event, the reactor protection system actuates on high RCS pressure. The staff reviewed the submittal and the analyses described in the licensee's UFSAR, Chapter 14, including the Anticipated Transient Without SCRAM (ATWS) event. The MSSVs function to limit the pressure increase so that the pressure does not exceed the applicable allowable values of the main steam piping and OTSGs. The turbine trip without ARTS event results in the energy from the reactor being transferred to the OTSGs, and with the nonsafety-related turbine bypass valves and atmospheric discharge valves not available, the sole release path would be the MSSVs, supporting the licensee's determination of this event as the limiting transient. Removing MSSVs from service reduces the rate that the steam/energy is released, resulting in an increase of pressure in the OTSGs and main steam lines. The NRC staff reviewed the ATWS event as well. Since the ATWS analyses utilize more realistic assumptions than the standard safety analyses, the staff agrees with the licensee that the turbine trip event analyzed is bounding in terms of secondary pressure response. Therefore, the staff agrees that the licensee's assessment of the limiting secondary side pressurization event is correct.

The licensee ran computer simulations to compare the severity of the affects of different combinations of MSSVs inoperable on each of the main steam lines. The licensee stated that the most limiting case for 100% power was two MSSVs inoperable on steam line 4. The NRC staff could not find an adequate explanation of how the licensee determined the most limiting case, between steam lines 2 and 4, in the application dated September 24, 2010. In a letter dated April 21, 2011, the licensee explained that they used sensitivity cases to determine the most limiting case was for inoperable MSSVs on the steam line with lesser MSSVs. The licensee provided a supplementary response in a letter dated May 27, 2011, showing the results of two MSSVs inoperable on steam line 2 for comparison with steam line 4. Also, the licensee's sensitivity studies found the most limiting individual MSSVs were the ones with the lowest lift setpoint. The licensee's case studies showed that the MSSVs with lower setpoint being inoperable have the most effect. Two very small MSSVs are installed on steam lines 1 and 3, and have a lift setpoint of 1040 psig; however, these valves have very limited relief capability; therefore, the licensee did not select these two low lift setpoint valves in their assessment of a limiting condition.

The NRC staff reviewed the data provided by the licensee, and compared the affects of having a valve inoperable with a lower flow rate with a lower lift setpoint, to a valve inoperable with a higher flow capacity with a higher lift setpoint. Based on that review, the NRC staff concluded that even though the valve with the larger flow capacity would relieve more steam, the valve with the lower lift setpoint opens sooner in the event, allowing it to relieve more energy out of the steam lines over the same time period. Therefore, the staff agrees that the licensee has properly determined the most limiting combination of inoperable valves.

The NRC staff reviewed the assumptions and limitations that the licensee incorporated into the calculation supporting inoperability of the MSSVs. In the evaluation, the licensee does not take credit for operation of the nonsafety-related turbine bypass valve and the atmospheric dump valves to relieve system pressure. This assumption is in alignment with the staff's position that does not allow credit for mitigation by nonsafety-related equipment that is not designed to be available during design-basis accidents. Therefore, the staff concurs with the licensee's assessment that the nonsafety-related turbine bypass valve and the atmospheric dump valves will not be modeled for accident mitigation. Hence, the MSSVs become the sole mechanism for relieving pressure from the main steam lines.

The licensee's calculation provides a graphical representation of the critical parameters of the primary and secondary systems. In the original application dated September 24, 2010, the licensee's graphs representing the pressure in the main steam lines that the MSSVs opened did not appear to align with the valve's lift setpoint. The MSSVs are designed to begin opening at a specific design pressure; however, there is a degree of allowable drift. In addition, the valves do not go immediately full open; the valves gradually open to full flow conditions. The staff requested that the licensee address how they accommodated the valves' design features and system piping configurations into their calculations. By letter dated April 21, 2011, the licensee explained that their analyses model the MSSVs to lift at 3% above their nominal setpoint. The licensee also explained that the model partially opens the valves at 103% of the setpoint, and ramps to the full open position at 106%. The licensee explained that their model includes a pressure drop from system piping, which reflects the difference in the lift setpoint pressure and the point depicted on the graph where the MSSVs lifts. The staff reviewed the licensee's assumptions and modeling inputs, and concurs that the model has allowances for the 3%

setpoint tolerance allowed by the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) for MSSVs. The staff also finds that the incorporation of ramp opening to 106% into the modeling process adequately accounts for a 3% accumulation. Therefore, the staff finds the licensee has adequately modeled the physical valve lift characteristics and piping pressure drop, in order to produce a conservative assessment of MSSV behavior.

The reactor protection system is normally designed to provide a reactor trip on overpower at a setpoint of 105.1% in order to provide assurance that the power will not exceed the safety analysis limit of 112%. The function of the 105.1% overpower trip is to prevent fuel damage in a reactivity transient and is not specifically oriented to protect against excessive overpressurization of the secondary side. The licensee conservatively modeled a 2% instrument uncertainty into the initial power level of the transient analysis (102%). Upon a turbine trip, the steam pressure in the main steam lines is relieved by lifting of the MSSVs at staggered pressures. The pressure rise in the RCS from the lost of normal heat removal by the OTSGs is mitigated first by lifting of the pressurizer power operated relief valve and then by opening both pressurizer safety valves. On the secondary side, once the peak OTSG pressure has been reached, the saturation pressure of the secondary side of the OTSG is determined by the lowest MSSV lift setpoint. If the five MSSVs with the lowest setpoints are inoperable, the remaining MSSV (lowest setpoint remaining of 1080 psig) will cool the reactor coolant cold leg to a temperature of approximately 555°F. Based on B&W natural circulation tests, an approximate 50°F core delta temperature would result in a core exit temperature of approximately 605°F. For this temperature, adequate subcooling in the RCS is assured above 1800 psi. Therefore, in accordance with GDC-10 and GDC-15, the staff is assured that the design conditions of the reactor coolant pressure boundary and the fuel will not be exceeded during any condition of normal operation, including anticipated operational occurrences with the changes proposed.

The NRC staff notes that the application dated September 24, 2010, contained an AREVA calculation which provided a justification for increasing the lift setpoint for MSSV MS-V-21A and MS-V-21B from 1040 to 1050 psig. The licensee has opted not to implement this change. Therefore, the staff did not review the proposed lift setpoint change.

3.3.2 Evaluation at Full Power (102% power, 2772 MWth)

The licensee performed a scoping analysis at full power to show the effects on the main steam line pressure and the steam generator pressure with two MSSVs inoperable. The licensee provided the results of their analysis in the submittal dated September 24, 2010, Table 4-1. The maximum RCS pressure reached was 2581.11 psia, where the limiting pressure to remain bounded by the startup accident is 2707 psia. The maximum OTSG pressure was 1170.97 psia, as compared to a maximum allowed pressure of 1279.7 psia, and the maximum steam line pressure was 1159.06 psia, less than the allowable value of 1169.7 psia. Therefore, the licensee concludes that in the event of a turbine trip transient with two MSSVs inoperable, the pressure in RCS, OTSGs, and main steam lines will not exceed their specified limits.

The staff reviewed the graphical representations of the main parameters of the primary and secondary systems at full power conditions. The graphs were produced using an acceptable NRC code. The assumptions, inputs, and limitations used in the code were conservative. The graphs show the primary and secondary parameters remain within the defined acceptance criteria provided by the ASME Code and the licensee's design and licensing basis. Therefore, the staff finds the licensee's analysis of allowing two inoperable MSSVs on each of the two OTSGs at full power conditions to be acceptable.

3.3.3 Evaluation at Reduced Power (92% of 2772 MWth)

The licensee performed a scoping analysis to show the effects on the main steam line pressure and the steam generator pressure with three MSSVs inoperable at a reduced power of 92%. The licensee provided the results of their analysis in the submittal dated September 24, 2010, Table 5-1. The maximum RCS pressure reached was 2578.82 psia, compared to the startup accident pressure of 2707 psia. The maximum OTSG pressure was 1174.42 psia, compared to the limit of 1279.7 psia. The maximum steam line pressure was 1165.19 psia, compared to the limit of 1169.7 psia. Therefore, the licensee concludes that in the event of a turbine trip transient with three MSSVs inoperable at a reduced power of 92%, the pressure in RCS, OTSGs, and main steam lines will not exceed their allowable values.

The NRC staff reviewed the graphical representations of the main parameters of the primary and secondary systems at 92% power. The graphs were produced using an acceptable NRC code. The assumptions, inputs, and limitations used in the code were conservative. The graphs show the primary and secondary parameters remain within the defined acceptance criteria provided by the ASME Code and the licensee's design basis. Therefore, the staff finds the licensee's analysis of allowing three inoperable MSSVs on each of the two OTSGs at or below 92% power to be acceptable.

3.3.4 Evaluation at Reduced Power (62% of 2772 MWth)

The licensee performed a scoping analysis to show the effects on the main steam line pressure and the steam generator pressure with five MSSVs inoperable at a reduced power of 62%. The licensee provided the results of their analysis in Table 5-4 found in Attachment to letter dated May 27, 2011. The maximum RCS pressure reached was 2598.09 psia, as compared to the maximum startup accident limit of 2707 psia. The maximum OTSG pressure was 1169.01 psia, as compared to the limit of 1279.7 psia. The maximum steam line pressure was 1164.94 psia, as compared to the limit of 1169.7 psia. Therefore, the licensee concludes that in the event of a turbine trip transient with five MSSVs inoperable at a reduced power of 62%, the pressure in RCS, OTSGs, and main steam lines will not exceed the established allowable values.

The staff reviewed the graphical representations of the main parameters of the primary and secondary systems at 62% power. The graphs were produced using an acceptable NRC code. The assumptions, inputs, and limitations used in the code were conservative. The graphs show the primary and secondary parameters remain within the defined acceptance criteria provided by the ASME Code and the licensee's design basis. Therefore, the staff finds the licensee's analysis of allowing five inoperable MSSVs on each of the two OTSGs at or below 62% power conditions to be acceptable.

3.3.5 Evaluation at Reduced Power (77% of 2772 MWth)

The licensee proposed that TMI-1 be allowed to operate at a power level of 77% with four MSSVs inoperable per steam generator. However, in the submittal dated September 24, 2010, the licensee did not provide a supporting analysis at 77% power. The staff requested that the licensee provide a basis for allowing this number of MSSV's to inoperable at 77% power. By letter dated April 21, 2011, the licensee explained that the basis for the number of MSSVs allowed inoperable at 77% was based upon a linear relationship between the peak pressures and reactor power. Specifically, the licensee calculated the maximum pressure at 92% power with four MSSVs inoperable and at 62% power with four MSSVs inoperable. Using a linear relationship, the licensee then calculated the reactor power level, with four MSSVs inoperable, where the peak steam line pressure will not exceed the maximum allowable pressure. This power level was calculated to be 78%. Therefore, the licensee proposed that with four MSSVs inoperable, the reactor power must be reduced to less than 77% power.

In addition, by letter dated May 27, 2011, the licensee provided the results of an explicit analysis using the NRC-approved R5/M2-BW method, confirming that, with four MSSVs inoperable at an initial reactor power level of 77%, for the analyzed event, maximum RCS pressure, SG pressure and steam line pressure will be within allowable values. The NRC staff reviewed the licensee's explicit analyses and concludes that the proposed technical specification permitting plant operation at 77% power with up to four MSSVs inoperable per steam generator is acceptable, because the licensee performed the analysis in accordance with NRC-approved methodology and acceptable results were obtained.

3.3.6 Evaluation of Allowing Inoperable MSSVs on each OTSG

In the application dated September 24, 2010, the licensee refers to an AREVA calculation (reference 11) that demonstrates "that the number of MSSVs out of service on one steam generator does not affect the number of MSSVs that can be out of service on the other steam generator." In Table 4-3, of the submittal dated September 24, 2010, the licensee provided the results of a sensitivity analysis at full power with three MSSVs inoperable on one and on both steam generators. The licensee concluded that the maximum steam pressure reached in the RCS, OTSGs, and main steam lines were approximately the same in all of these cases ('A' SG, 'B' SG and 'A+B' SG MSSVs inoperable).

The NRC staff reviewed the data provided in the licensee's calculation of the main parameters of the primary and secondary systems for MSSVs inoperable on one OTSG and the same number of inoperable MSSVs on both OTSGs. The data was produced using an acceptable NRC code. The assumptions, inputs, and limitations used in the code were conservative. The data shows the primary and secondary parameters were approximately equal; hence, the number of inoperable MSSVs on one OTSG does not significantly affect the performance of the other OTSG. The staff finds the licensee's analysis allowing the same number of inoperable MSSVs on each of the two OTSGs at full and reduced power conditions to be acceptable. In addition, the NRC staff notes that allowances for inoperable MSSVs on each OTSG aligns with the guidance provided in NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants."

3.3.7 Evaluation of Overpower Trip Setpoints

In order to provide assurance that the maximum reactor power will not be exceeded, the licensee has proposed a reduction in the overpower setpoint when more than two MSSVs are inoperable on one steam generator. The licensee will preserve the overpower trip setpoint uncertainty of 6.9%, which is actually the difference between the maximum reactor power for protection of the specified acceptable fuel design limit (112%) and the high flux trip setpoint (105.1%). Stated differently, this difference is the power overshoot associated with the limiting departure from nucleate boiling ratio transient on which the high neutron flux trip setpoint is based.

The licensee clarified by letter dated March 18, 2011, that the 6.9% full power uncertainty applied to the overpower trip setpoints at each of the analyzed power levels is actually more conservative than necessary because it is not necessarily applicable to the turbine trip transient.

The licensee stated further that the applicable power uncertainty is 2%, which corresponds to the reactor power measurement uncertainty.

By setting the trip setpoints 6.9% below the analyzed power levels, the licensee ensures that the analysis is conservative. This is because if operating at a higher power level, between the loss of heat sink and the reactor trip, the reactor will add more energy to the coolant. If the trip setpoint is set at a lower power level than analyzed, the reactor will add less energy to the coolant, and ultimately to the main steam system. Because setting the trip setpoint lower than the analyzed power level (with uncertainty) is a conservative analytic approach, the NRC staff finds the proposed setpoints acceptable.

3.4 Conclusion

The NRC staff has reviewed the licensee's proposed change to TS 3.4.1.2.3 to allow inoperable MSSVs during power operation and the supporting calculations. The staff finds that the proposed TS is in accordance with guidance provided in applicable codes and standards and is therefore acceptable. The staff agrees that the main steam system will remain fully capable of providing its design safety function of overpressure protection as allowed by the revised TS, at the reactor power levels specified, with the appropriate number of MSSVs operable.

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. 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 (75 FR 74096). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: S. Gardocki
B. Parks
P. Bamford

Date: September 14, 2011

September 14, 2011

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: MAXIMUM ALLOWABLE POWER WITH INOPERABLE MAIN STEAM SAFETY VALVES (TAC NO. ME4808)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 277 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 September 24, 2010,[1] as supplemented by additional letters.[2]

The amendment revises Technical Specification 3.4.1.2.3, to allow up to two Main Steam Safety Valves (MSSVs) per steam generator to be inoperable with no required reduction in power level. It also revises the required maximum overpower trip setpoints for any additional inoperable MSSVs consistent with the plant transient analysis. The change requires that with less than four MSSVs associated with either steam generator operable, the plant would be required to be brought to the hot shutdown condition.

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. 277 to DPR-50
2. Safety Evaluation
cc: Distribution via Listserv

DISTRIBUTION
PUBLIC
RidsAcrsAcnw_MailCTR Resource
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RidsNrrDorlLpl1-2 Resource
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LPLI-2 R/F
BParks, NRR
RidsOgcRp Resource
SGardocki, NRR
AGuzzetta, NRR

ADAMS Accession No.: ML112360581 * via memo **via email

OFFICE	LPL1-2/PM	LPLI-2/LA	SRXB/BC	SBPB/BC	ITSB/BC	OGC	LPL1-2/BC
NAME	PBamford	ABaxter **	TUlses *	GCasto *	RElliott	NLO MSpencer	HChernoff
DATE	08/25/2011	08/29/2011	08/19/2011	08/15/2011	08/30/2011	09/07/2011	09/14/2011

Official Record Copy

1. Agencywide Documents Access and Management System (ADAMS) Accession No. ML102780570.
2. Letters dated March 18, 2011 (ADAMS Accession No. ML110770296), April 21, 2011 (ADAMS Accession No. ML111120003), and May 27, 2011 (ADAMS Accession No. ML111920354).