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

July 28, 2015

Mr. Bryan C. Hanson
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: TECHNICAL SPECIFICATIONS TO MODIFY REACTOR COOLANT SYSTEM PRESSURE ISOLATION CHECK VALVE MAXIMUM ALLOWABLE LEAKAGE LIMITS (TAC NO. MF5108)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment No. 286 to Renewed Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated October 30, 2014, as supplemented by letter dated June 10, 2015.

The amendment revises the TMI-1, Technical Specification Table 3.1.6.1, "Pressure Isolation Check Valves Between the Primary Coolant System & LPIS [Low Pressure Injection System]," maximum allowable leakage limits.

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,

Robert L. Gladney

Robert L. Gladney, 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. 286 to Renewed DPR-50
2. Safety Evaluation

cc w/enclosures: 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 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 286
Renewed License No. DPR-50

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated October 30, 2014, as supplemented by letter dated June 10, 2015, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 286, 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 as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications
and Facility Operating License

Date of Issuance: July 28, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 286

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following page of the 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
4	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 area of change.

Remove	Insert
3-15b	3-15b

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 286, 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.

TABLE 3.1.6.1

PRESSURE ISOLATION CHECK VALVES BETWEEN
THE PRIMARY COOLANT SYSTEM & LPIS

System	Valve No.	Maximum Allowable Leakage
Low Pressure Injection		
Train A	CF-V5A	≤5.0 GPM
	DH-V22A	≤5.0 GPM
Train B	CF-V5B	≤5.0 GPM
	DH-V22B	≤5.0 GPM

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 286

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-50

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated October 30, 2014, as supplemented by letter dated June 10, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14304A083 and ML15161A333, respectively), Exelon Generation Company, LLC (Exelon, the licensee), requested changes to the Technical Specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1).

The proposed amendment would revise the TMI-1 TS Table 3.1.6.1, "Pressure Isolation Check Valves Between the Primary Coolant System & LPIS [Low Pressure Injection System]," by deleting footnotes (a)1 through (a)4 involving additional restrictions to the maximum allowable leakage limits of 5 gallons per minute (gpm).

The supplement dated June 10, 2015, 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 consideration determination as published in the *Federal Register* on December 9, 2014 (79 FR 73110).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance that the NRC staff considered in its review of the proposed TS change are provided below.

- Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," establishes the fundamental regulatory requirements. Specifically, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. Appendix A did not exist when TMI-1 was built. 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 "AEC GDC"). The AEC published the final rule that added Appendix A in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "GDC"). Differences between the AEC GDC and 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. TMI-1, plant-specific principal design criteria are described in Chapter 1 of the Updated Final Safety Analysis Report (UFSAR). In the following sections, the similar AEC GDC is noted in parentheses ( ) after the Appendix A requirement.

- General Design Criterion 14, "Reactor Coolant Pressure Boundary," requires that:

  > The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

  (AEC GDC Criterion 9, "Reactor Coolant Pressure Boundary (Category A)," requires that "the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.")

- The licensee stated that General Design Criterion 54, "Piping Systems Penetrating Containment," is applicable, which requires that:

  > Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

  (The NRC staff reviewed GDC 54 and determined that it meets the intent of AEC GDC 51 and AEC GDC 57. AEC GDC 51, "Reactor Coolant Pressure Boundary Outside Containment (Category A)," requires if part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, shall include consideration of the environmental and population conditions surrounding the site. AEC GDC 57, "Provisions for Testing of Isolation Valves (Category A)," requires that capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.)

- The licensee stated that General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment," is applicable, which requires:

  Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

  (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
  (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
  (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
  (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

  Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

  Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

  (The NRC staff reviewed GDC 55 and determined that it meets the intent of AEC GDC 51. AEC GDC 51, "Reactor Coolant Pressure Boundary Outside Containment (Category A)," requires if part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, shall include consideration of the environmental and population conditions surrounding the site.)

- 10 CFR 50.36(c)(2)(ii)(C), "Criterion 3," requires, in part, that:

  A [TS] limiting condition for operation of a nuclear reactor must be established for...[a] structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- 10 CFR 50.36(c)(3), "Surveillance requirements," requires:

  Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

- 10 CFR 50.55a(f)(4), "Inservice testing standards requirement for operating plants," requires, in part, that:

  . . . pumps and valves that are classified as [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] ASME Code Class 1, Class 2, and Class 3 must meet the inservice test requirements, (except design and access provisions) set forth in the ASME OM Code [Code for Operation and Maintenance of Nuclear Power Plants] and addenda.

3.0 TECHNICAL EVALUATION

The licensee proposed to revise TMI-1 TS to provide requirements that are more consistent with the industry standard in accordance with NUREG-1430, "Standard Technical Specifications-Babcock and Wilcox Plants," Revision 4, with respect to Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage and to correct two typographical errors and one clerical error.

The licensee stated, in part, that:

> *The proposed changes will revise the Technical Specifications (TS) Table 3.1.6.1, "PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS," by deleting footnotes (a)1 through (a)4 involving additional restrictions to the Maximum Allowable Leakage limits of 5 gallons per minute (gpm). In addition, the proposed changes will correct two typographical errors and one clerical error.*
>
> *Currently, the Maximum Allowable Leakage limit for these Reactor Coolant System Pressure Isolation Valves (RCS PIVs) is ≤ 5.0 gpm. The Maximum Allowable Leakage limit is further restricted with Footnote (a) that describes additional incremental surveillance testing acceptance criteria prior to reaching the 5 gpm limit, which is based in part on the remaining margin between the 5 gpm limit and the last measured leakage. This proposal requests deletion of the Footnotes (a)1 through (a)4, making it consistent with Improved Standard Technical Specification (ITS) surveillance requirements (NUREG 1430 SR 3.4.14.1).*
>
> *Technical Specification Changes*
>
> *TS Table 3.1.6.1, "PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS," Maximum Allowable Leakage Footnotes (a)1 through (a)4 below will be deleted:*

Footnote:

(a)

1. *Leakage rates less than or equal to 1.0 gpm are considered acceptable.*

2. *Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.*

3. *Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.*

4. *Leakage rates greater than 5.0 gpm are considered unacceptable.*

The following [were believed to have been] typographical errors [needing correction]:

1. *Table 3.1.6.1, Train A, CF-VSA has a typographical error and is being corrected to CF-V5A.*

2. *In TS 3.1.6.10.a, the acronym LIPS is being corrected to LPIS.*

The following clerical error is being corrected:

1. *In Table 3.1.6.1 delete "(≤5.0 GPM for all valves)" phrase that is on the same typing line that begins with "Low Pressure Injection."*

2. *In table 3.1.6.1 revise valve list to read:*

CF-V5A	*≤ 5.0 GPM*
DH-V22A	*≤ 5.0 GPM*
CF-V5B	*≤ 5.0 GPM*
DH-V22B	*≤ 5.0 GPM*

[Licensee] Technical Evaluation

PIV leakage testing was originally established by the NRC in response to concerns regarding the intersystem loss-of-coolant accident (ISLOCA), which was identified in the Reactor Safety Study of 1975, WASH-1400 [...]). An ISLOCA event at TMI would involve the failure of two in-series RCS PIVs, which would subject a low pressure system outside of containment to full primary coolant system pressure. The low pressure system would consequently rupture, *resulting in a [loss-of-coolant accident] LOCA that would bypass containment, thereby jeopardizing the ability for long term reactor core cooling. NUREG-0103 Rev 4 Standard Technical Specifications for Babcock and Wilcox [Pressurized Water Reactor] PWRs (Fall 1980) initially limited RCS PIV leakage to 1 gpm.*

TS LCO 3.1.6.10 and TS Table 3.1.6.1 provide requirements for the Maximum Allowable Leakage for RCS PIVs, including the limiting condition for operation, action requirements and surveillance requirements. TS Table 3.1.6.1 limits Maximum Allowable Leakage for each RCS PIV to ≤ 5 gpm. For TMI, the additional leakage surveillance testing acceptance criteria embodied in Table 3.1.6.1 Footnote (a) evolved from NRC TMI Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, dated April 20, 1981 [ADAMS Accession Nos. ML003764606, ML003764617, ML003764625, and ML003764633]. These additional requirements provided further attention to the rate of valve leakage degradation in the 1 to 5 gpm range.

Although this specification provides a limit on allowable RCS PIV leakage, and additional rate of valve leakage degradation limit, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The Maximum Allowable Leakage limit is an indication that the PIVs between the RCS and the connecting lower pressure system are degraded or degrading. Excessive RCS PIV leakage could lead to overpressure of the low pressure piping or components, potentially resulting in an ISLOCA outside of containment.

The NRC, through its approval of Babcock and Wilcox Improved Standard Technical Specifications (ITS), NUREG-1430, has revised the leakage rates enforced in the TMI April 20, 1981 Order and endorsed an RCS PIV leakage rate limit without additional incremental surveillance testing acceptance criteria (i.e., Footnotes (a)1 through (a)4). The revised ITS RCS PIV leakage limit is 0.5 gpm per nominal inch of valve diameter with a maximum limit of 5 gpm. This change tightened the leakage requirement for smaller valves and relaxed it for larger ones. CF-V5A, B are nominal 14-inch diameter valves and the DH-V22A, B are nominal 10-inch diameter valves, and therefore have a Maximum Allowable Leakage rate of 5 gpm using ITS criteria. The propose[d] deletion of the incremental surveillance testing acceptance criteria is consistent with ITS, which contains no required additional incremental surveillance testing acceptance criteria as valve leakage progresses to the maximum allowable leakage of 5 gpm.

NUREG-0677, ["The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes,"] May 1980, evaluated various RCS PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the RCS PIVs can substantially reduce the probability of an ISLOCA. There was no consideration of additional incremental surveillance testing acceptance criteria prior to reaching a predetermined maximum allowable leakage in the probability calculations for a check valve leak failure and rupture failure calculations for the specific *configuration of two series check valves. Therefore, the probability of an ISLOCA is unaffected.*

The proposed change does not affect the current Inservice Testing (IST) Program. The RCS PIVs are Class 1 Category A/C valves and are tested in accordance with ASME OM Code, Edition 2004 through the 2006 addenda, which is the latest edition incorporated by reference in Paragraph (b) of 10 CFR 50.55a for TMI. ASME OM Code ISTC-3630, "Leakage Rate for Other Than Containment Isolation Valves," requires testing every 2 years. TMI TS require leakage testing every refuel outage or 9 months for forced outages. The requirements for water leakage rates are 0.5D gal/min or 5 gal/min, whichever is less, at function pressure differential, where D is nominal valve size in inches. TMI TS require the Maximum Allowable Leakage to be ≤ 5.0 gpm.
A potential concern related to deleting the PIV allowable leakage rates incremental surveillance testing requirements is that the low pressure systems isolated by the PIVs may not have sufficient pressure relief capacity to cope with the incremental increased leakage (up to 5 gpm limit). In-leakage exceeding the pressure relief capacity of a low pressure system could lead to its overpressurization and rupture. The maximum allowable leakage of 5 gpm remains the same and is well within the pressure relief valve capacity of the Core Flooding and Decay Heat Removal Systems. These relief valves are tested in accordance with ASME [American Society of Mechanical Engineers] O&M Code [Code for Operation and Maintenance of Nuclear Power Plants, or OM Code].

NRC Evaluation

The licensee's application stated that there were two typographical errors that needed correction. However, as part of the review, it was determined that the controlled copy of the TMI-1 TS did not contain these errors. Following a discussion with the licensee, the licensee submitted a supplemental letter indicating that the errors existed in the Exelon Electronic Document Management System (EDMS), but not on the controlled copies of the TMI-1 TS. Therefore, the proposed correction of typographical errors in TS 3.1.6.10.a (LIPS) and TS Table 3.1.6.1 (CF-VSA) was withdrawn. The licensee indicated that errors in the EDMS file have been entered into Exelon's Corrective Action Program for correction. The licensee also removed the proposed changes to TS page 3-13 from their submittal.

Pressure isolation valves are defined as two valves in series within the reactor coolant pressure boundary that separate the high-pressure RCS from an attached lower pressure system. Failure of a PIV could result in an over-pressurization event that could lead to a system rupture and possible release of fission products to the environment. Periodic leak rate testing of PIVs monitors overall valve health and reduces the probability of an over-pressurization event. Requirements for periodic leak rate testing of PIVs are specified in the ASME OM Code subsection ISTC, "Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants."

In October 1975, the NRC published WASH-1400, "Reactor Safety Study (RSS): An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (NUREG-75/014)." The RSS study identified in a PWR that an ISLOCA is a significant contributor to risk of core melt accidents (Event V). The design examined in the RSS study contained in-series check valves isolating high-pressure Primary Coolant System (PCS) from the LPIS piping. Failure of these check valves to function as a pressure isolation barrier could lead to the Event V accident.

In order to better define the Event V concern, all light water reactor (LWR) licensees were requested by Generic Letter 80-14, "LWR Primary Coolant System Pressure Isolation Valves," dated February 23, 1980, to review their present design to determine if an Event V configuration exists. TMI-1 responded to the generic letter and committed valves CF-VF5A/B and DH-V22A/B to the ordered maximum allowable leakage rate criteria of:

- Leakage rates less than or equal to 1.0 gpm are considered acceptable
- Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
- Leakage rates greater than 5.0 gpm are considered unacceptable.

The NRC staff recognized that the one gpm acceptance criteria may not be an indicator of imminent accelerated deterioration of valves or potential valve failure. The staff had a study prepared on the subject of leak test requirements for pressure isolation valves, "Inservice Leak Testing of Primary Pressure Isolation Valves, R. A. Livingston, EGG-NTAP-6175, February 1983, FIN A6367." As part of the study, a survey was conducted of nine licensees in order to obtain information regarding actual in-plant experience with the one gpm and Event V leak test criteria. The outcome of the study concluded that an allowable leak rate criteria of one-half gpm for each inch of nominal valve size up to a maximum of 5 gpm should be adopted with the understanding that the additional leakage requirements of ASME OM Code are still applicable. The choice of 5 gpm as an absolute maximum allowable leakage rate is based on the experience gained from the plants using this standard in compliance with the Event V orders.

The revised standard became effective in the mid-1980's and applied to all PWR and Boiling-Water Reactor designs. This change tightened the leakage requirement for smaller valves and relaxed it for larger ones. The higher leakage criteria decreased unnecessary valve maintenance and reduced overall personnel radiation exposure. Applicants and licensees may adopt the new test leak criteria in lieu of their existing commitments at their option. The revised PIV leakage requirement has been documented into NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants."

A potential concern related to increasing PIV allowable leakage rates is that the low pressure systems isolated by the PIVs may not have sufficient pressure relief capacity to cope with the increased leakage. Leakage exceeding the pressure relief capacity of a low pressure system would lead to its over pressurization and rupture. The licensee has stated in its submittal that the low pressure systems affected by the proposed TS changes have pressure relief valves installed and are capable of protecting the system from an over-pressurization event due to excessive PIV leakage. The NRC staff completed a review of the licensee's system configuration, installed components, inservice testing program plan, and the maintenance history. The review confirms that pressure relief valves are installed and are capable of meeting the revised leakage criteria. The NRC staff also confirmed that the valves are qualified and tested to the requirements of the ASME OM Code and have an acceptable maintenance leakage history.

The NRC staff has reviewed the license amendment request by letter dated October 30, 2014 (ADAMS Accession No. ML14304A083), as supplemented by letter dated June 10, 2015 (ADAMS Accession No. ML15161A333), and finds the proposed TS change to revise the leakage requirements for PIVs is acceptable on the basis that the changes are consistent with the requirements of the ASME OM Code and the recommendations in NUREG-1430, Revision 4, for Babcock and Wilcox designed plants, which were subject to NRC evaluation prior to issuance of the NUREG. The proposed change in TS will continue to assure that excessive PIV leakage is properly identified and resolved. In addition, the change will reduce unnecessary occupational radiation exposure by reducing PIV maintenance activities that do not significantly enhance the function of the PIVs. Based on the findings, the NRC staff concludes that there is reasonable assurance that the requirements specified in Section 2.0 of this safety evaluation will continue to be met. Therefore, the staff finds the proposed change acceptable.

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 changes surveillance requirements. 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 (79 FR 73110, December 9, 2014). 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 Contributor: M. Farnan

Date: July 28, 2015

July 28, 2015

Mr. Bryan C. Hanson
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: TECHNICAL SPECIFICATIONS TO MODIFY REACTOR COOLANT SYSTEM PRESSURE ISOLATION CHECK VALVE MAXIMUM ALLOWABLE LEAKAGE LIMITS (TAC NO. MF5108)

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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/
Robert L. Gladney, Project Manager
Plant Licensing Branch I-2
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Docket No. 50-289

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