January 7, 2015

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

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: MAIN STEAM LINE FLOW-HIGH ISOLATION RESPONSE TIME CHANGE (TAC NOS. MF3085 AND MF3086)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment Nos. 214 and 175 to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station, Units 1 and 2, respectively. These amendments consist of changes to the Technical Specifications (TSs) and Facility Operating Licenses in response to your application dated November 15, 2013, as supplemented by letters dated April 16, 2014, September 11, 2014, and November 7, 2014.

The amendments revise the TS requirements related to the response time for the main steam line flow-high isolation function.

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

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. Amendment No. 214 to Renewed NPF-39
2. Amendment No. 175 to Renewed NPF-85
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
Renewed License No. NPF-39

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated November 15, 2013, as supplemented by letters dated April 16, 2014, September 11, 2014, and November 7, 2014, 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. NPF-39 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 214, are hereby incorporated into this renewed license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report to reflect the revised main steam line break analysis.

FOR THE NUCLEAR REGULATORY COMMISSION

M Khanna

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

Attachment:
Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: January 7, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 214

RENEWED FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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 area of change.

Remove	Insert
Page 3	Page 3

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
3/4 3-23	3/4 3-23

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3515 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this renewed license shall be completed as specified. Attachment 1 is hereby incorporated into this renewed license.

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

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 214, are hereby incorporated into this renewed license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
1.	MAIN STEAM LINE ISOLATION	
	a. Reactor Vessel Water Level	
	1) Low, Low - Level 2	N.A.
	2) Low, Low, Low - Level 1	≤1.0###*
	b. DELETED	DELETED
	c. Main Steam Line Pressure - Low	≤1.0###*
	d. Main Steam Line Flow - High	≤1.0###*
	e. Condenser Vacuum - Low	N.A.
	f. Outboard MSIV Room Temperature - High	N.A.
	g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
	h. Manual Initiation	N.A.
2.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
	a. Reactor Vessel Water Level Low - Level 3	N.A.
	b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
	c. Manual Initiation	N.A.
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION	
	a. RWCS Δ Flow - High	N.A.##
	b. RWCS Area Temperature - High	N.A.
	c. RWCS Area Ventilation Δ Temperature - High	N.A.
	d. SLCS Initiation	N.A.
	e. Reactor Vessel Water Level - Low, Low - Level 2	N.A.
	f. Manual Initiation	N.A.

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
Renewed License No. NPF-85

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated November 15, 2013, as supplemented by letters dated April 16, 2014, September 11, 2014, and November 7, 2014, 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. NPF-85 is hereby amended to read as follows:

   2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 175, are hereby incorporated into this renewed license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report to reflect the revised main steam line break analysis.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena Khanna

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

Attachment:
Changes to the Technical Specifications
and Renewed Facility Operating License

Date of Issuance: January 7, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 175

RENEWED FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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 area of change.

Remove	Insert
Page 3	Page 3

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
3/4 3-23	3/4 3-23

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels of 3515 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 175, are hereby incorporated into this renewed license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
1.	MAIN STEAM LINE ISOLATION	
	a. Reactor Vessel Water Level	
	1) Low, Low - Level 2	N.A.
	2) Low, Low, Low - Level 1	≤1.0###*
	b. DELETED	DELETED
	c. Main Steam Line Pressure - Low	≤1.0###*
	d. Main Steam Line Flow - High	≤1.0###*
	e. Condenser Vacuum - Low	N.A.
	f. Outboard MSIV Room Temperature - High	N.A.
	g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
	h. Manual Initiation	N.A.
2.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
	a. Reactor Vessel Water Level Low - Level 3	N.A.
	b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
	c. Manual Initiation	N.A.
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION	
	a. RWCS Δ Flow - High	N.A.##
	b. RWCS Area Temperature - High	N.A.
	c. RWCS Area Ventilation Δ Temperature - High	N.A.
	d. SLCS Initiation	N.A.
	e. Reactor Vessel Water Level - Low, Low - Level 2	N.A.
	f. Manual Initiation	N.A.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 214 AND 175

TO RENEWED FACILITY OPERATING LICENSE NOS. NPF-39 AND NPF-85

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By application dated November 15, 2013, as supplemented by letters dated April 16, 2014, September 11, 2014, and November 7, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML13322A448, ML14106A642, ML14255A088, and ML14311A892, respectively), Exelon Generation Company, LLC (Exelon, the licensee), requested changes to the Technical Specifications (TSs) for Limerick Generating Station (LGS), Units 1 and 2.

The proposed amendment would revise the TS requirements related to the response time for the main steam line (MSL) flow-high isolation function. Specifically, the amendment would revise TS Table 3.3.2-3, "Isolation System Instrumentation Response Time," for trip function 1.d, "Main Steam Line Flow - High," to change the response time from ≤ 0.5 seconds to ≤ 1.0 seconds.

The licensee's application stated that the purpose of this request was to address a decrease in response time testing margin identified during surveillance testing of the relay logic that initiates isolation of the MSL on high flow. The application indicated that the proposed change is based on using available design basis margin in the main steam line break (MSLB) accident analysis.

The supplements dated April 16, 2014, September 11, 2014, and November 7, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 4, 2014 (79 FR 6642).

2.0 REGULATORY EVALUATION

The NRC staff review took into consideration the regulatory requirements and guidance documents discussed below.

The NRC's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TSs include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) Limiting Conditions of Operation (LCOs); (3) Surveillance Requirements (SRs); (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

As discussed in 10 CFR 50.36(c)(2), LCOs are the lowest functional capability or performance level of equipment required for safe operation of the facility. When LCOs are not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met.

As discussed in 10 CFR 50.36(c)(3), SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems 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.

Appendix A to 10 CFR Part 50, GDC 13, "Instrumentation and control," requires, in part, that instrumentation be provided to monitor variables and systems, and that controls are provided to maintain these variables and systems within prescribed operating ranges.

Appendix A to 10 CFR Part 50, GDC 15, "Reactor coolant system design," requires that the reactor coolant system and associated auxiliary, control, and protection systems 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.

Appendix A to 10 CFR Part 50, GDC 19, "Control room," requires, in part, that adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 roentgen equivalent man (rem) whole body, or its equivalent to any part of the body, for the duration of the accident.

Appendix A to 10 CFR Part 50, GDC 20, "Protection system functions," requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

Appendix A to 10 CFR Part 50, GDC 29, "Protection against anticipated operational occurrences," requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

As discussed in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," the emergency core cooling system (ECCS) must be designed so that its calculated performance, following a postulated loss-of-coolant accident (LOCA), conforms to the acceptance criteria specified in 10 CFR 50.46(b). As stated in 10 CFR 50.46(b)(1), "Peak cladding temperature," the calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

The requirements in 10 CFR 50.67, "Accident source term," provides a mechanism for power reactor licensees to voluntarily replace the traditional accident source term used in design-basis accident analyses with an "alternative source term" (AST). The NRC approved the AST methodology and radiological dose consequence analyses for design basis accidents (DBAs) for LGS, Units 1 and 2, via License Amendments 185 and 146 on August 23, 2006 (ADAMS Accession No. ML062210214).

NRC Standard Review Plan (SRP), NUREG-0800, Section 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) [boiling-water reactor]," Revision 2, dated July 1981 (ADAMS Accession No. ML052350151), provides guidance for NRC staff review of radiological doses, due to a MSLB outside containment, for a BWR.

3.0 TECHNICAL EVALUATION

3.1 Instrumentation Review

The NRC staff reviewed the proposed amendment from an instrumentation perspective as described in safety evaluation (SE) Sections 3.1.1, 3.1.2, and 3.1.3 below.

3.1.1 Background

LGS TS Table 3.3.2-3 provides response time requirements for the isolation system instrumentation. In accordance with LCO 3.3.2, the response times in this table must be met for the associated instrumentation channel to be operable. The response times in Table 3.3.2-3 are verified in accordance with SR 4.3.2.3. Currently, in accordance with Table 3.3.2-3 (trip function 1.d), the required instrument response time for the main steam isolation valve (MSIV) closure initiation instrument channel on high steam line flow is 500 milliseconds (0.5 seconds).

Each MSL flow element supplies 4 flow transmitters. Each transmitter inputs into each MSIV's closing circuit. The MSIV isolation logic is 1-out-of-2 taken twice.

TS Table 3.3.2-3 was revised in LGS, Units 1 and 2, License Amendments 132 and 93 in December 1998 (ADAMS Accession No. ML011560614), to eliminate response time testing of the flow transmitter in each MSL flow instrument channel. As a result, the MSIV closure initiation time of 500 milliseconds was allocated as follows: (1) 355 milliseconds for the transmitter response time; and (2) 145 milliseconds for the TS-required limit for the trip unit and the three relays of the instrument channel. The licensee established an administrative limit (allotted time) of 135 milliseconds to proactively identify for drift, and hence alert if the TS limit was being approached.

In 2012, the licensee reported several occurrences in which the response time for the MSL high flow instrument channels exceeded the TS limit of less than or equal to 0.5 seconds during performance of surveillance tests. The licensee submitted Licensee Event Report 2012-008-00 to document these occurrences (ADAMS Accession No. ML12318A120).

As discussed in the licensee's letter dated April 16, 2014 (ADAMS Accession No. ML14106A642), the response time surveillance testing for these instrument channels is performed every 2 years.

3.1.2 Response Time

As discussed in Attachment 1 of the application dated November 15, 2013, currently, the total response time for an MSL isolation on high steam flow is 5.5 seconds. The MSIVs start closing at 0.5 seconds on a high flow signal and are fully closed at 5.5 seconds. The proposed amendment would increase the instrument channel closure initiation from 0.5 seconds to 1.0 seconds. Consequently, the MSIVs would start closing at 1.0 seconds on a high flow signal and would be fully closed at 6.0 seconds.

The proposed amendment does not modify any setpoint values in the LGS TSs. Instead, the licensee is requesting a modification to the response time for MSIV closure initiation on MSL high flow. Therefore, the NRC's instrumentation review focused on the margin available for the instrumentation to provide its MSIV closure initiation function (i.e., focus is on the change from 0.5 seconds to 1.0 seconds). The reactor systems review, in SE Section 3.2 below, focused on the impact of the change from 5.5 seconds to 6.0 seconds for MSIV closure with respect to the accident analysis.

In the application and supplements, the licensee explained that it performed an evaluation of the response time for each element in the instrument channel using vendor data specifications. With this information, the licensee estimated the cumulative response time for the instrument channel (trip unit and three relays) to be equal to 199 milliseconds. Therefore, the current allocated administrative response time for the instrument channel of 135 milliseconds, as well as the allocated TS-required response time of 145 milliseconds, could not be satisfied. The licensee noted that, based on bench testing and engineering review of surveillance data, the challenge for the instrument channel to meet the response time requirements was not a result of equipment performance (e.g., increased drift) but rather a result of the limited margin available.

As discussed in the licensee's supplement dated April 16, 2014, the cumulative response time of 199 milliseconds would be used as the new administrative limit (allotted time) to proactively identity any equipment concerns (e.g., for drift), and hence alert if the TS limit is being approached. The allocated TS-required limit for the channel response time would be modified to account for the increase in the TS-required instrument response time for the MSIV closure initiation. As a result, the allotted time for the TS required limit for the trip unit and the isolation and implementation relays of the instrument channel increase from 0.145 seconds to 0.645

seconds. The following table summarizes the current design basis time frames versus the proposed design basis time frames.

Parameter	Current Design Basis	Proposed Design Basis
MSIVs start to close on high flow signal	0.5 seconds	1.0 seconds
MSIVs fully closed	5.5 seconds	6 seconds
Transmitter response time	0.355 seconds	0.355 seconds
Trip unit and relays response time (TS-required limit)	0.145 seconds (Basis: 0.5 seconds for MSIV starting to close minus 0.355 seconds for the transmitter response time.)	0.645 seconds (Basis: 1.0 seconds for MSIV starting to close minus 0.355 seconds for the transmitter response time.)
Trip unit and relays response time (administrative limit)	0.135 seconds	0.199 seconds

Based on the discussion and results described above, the proposed value for the trip function 1.d, "Main Steam Line Flow - High," in TS Table 3.3.2-3, "Isolation System Instrumentation Response Time," would be modified from ≤ 0.5 seconds to ≤ 1.0 seconds.

3.1.3 Instrumentation Conclusion

Based on the discussion above, the NRC staff concludes that the proposed change from 0.5 seconds to 1.0 seconds, for the MSIVs to start to close on a high flow signal, is reasonable given the response time of the individual components in the instrument channel. As such, the staff finds the proposed amendment to be acceptable from an instrumentation perspective.

3.2 Reactor Systems Review

The NRC staff reviewed the proposed amendment from a reactor systems perspective as described in SE Sections 3.2.1, 3.2.2, and 3.2.3 below. As noted above, the reactor systems review focused on the impact of the change from 5.5 seconds to 6.0 seconds for MSIV closure with respect to the accident analysis.

3.2.1 MSLB Analysis

An increase in the response time of the instrument channel that initiates closure of the MSIVs may result in an increase of total released mass for the postulated event, thereby potentially leading to an increase in the calculated radiological dose.

The MSLB analysis is described in Section 15.6.4 of the LGS Updated Final Safety Analysis Report (UFSAR). UFSAR Section 15.6.4.4 states that it is assumed that the MSIVs start to close at 0.5 seconds on a high flow signal and are fully closed at 5.5 seconds. This section of the UFSAR also states that, for the radiological consequence evaluation, a total mass of 140,000 pounds-mass (lbm) is assumed.

Attachment 4 to the application dated November 15, 2013, provided an analysis performed by GE Hitachi Nuclear Energy (GEH). The GEH analysis (0000-0158-9651-NP, Revision 0, dated October 2013) evaluated the change in the MSL mass flow resulting from increasing the MSIV closure initiation signal time from 0.5 seconds to 1.0 seconds.

The GEH analysis evaluated the current MSLB analysis assumptions found in Section 15.6.4.4 of the UFSAR. UFSAR Section 15.6.4.4.f currently contains the following assumption for the MSLB analysis:

> Level rise time is conservatively assumed to be 1 second. Mixture quality is conservatively taken to be a constant 7% (steam weight percentage) during mixture flow.

With respect to the above assumption, GEH stated that "[t]his assumption is not necessary, SAFER is a systems code and can calculate the time varying two phase break flow during the STMO [MSLB outside of containment] event."

In a request for additional information, the NRC staff requested clarification to support the statement "this assumption is not necessary." In the supplement dated September 11, 2014, the licensee clarified that the current method, for determining the LGS MSLB mass flow for dose analysis, is based on conservative simplifying assumptions as described in LGS UFSAR Section 15.6.4 and is not based on the SAFER evaluation model.

The proposed method for dose analysis, described in the GEH analysis, is based on analyses performed utilizing the NRC staff-approved SAFER 04A code/model. SAFER is a systems code that calculates the time varying reactor level response and two-phase mass flow during the MSLB event. In the supplement dated November 7, 2014, the licensee confirmed that SAFER 04A, used for MSLB outside containment analyses as documented in GE Nuclear Energy report NEDC-31270-P[1], is consistent with application of the SAFER methodology approved by the NRC staff in report NEDE-23785-1-P-A[2]. In the supplement dated November 7, 2014, the licensee confirmed that no exceptions were taken to the evaluation methodology or the applicable limitations and conditions in the NRC safety evaluation approving NEDE-23785-1-P-A.

The current simplifying assumptions in the LGS UFSAR do not apply to the proposed method for the MSLB analysis. No changes to the SAFER code were required to perform the analyses. As discussed in the supplement dated September 11, 2014, as part of the implementation of the proposed amendment, LGS UFSAR Section 15.6.4.4, including the currently listed assumptions,

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1 GE Nuclear Energy, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis for Limerick Generating Station, Units 1 and 2, NEDC-32170P, Revision 2, dated May 1995.
2 General Electric Company, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," NEDE-23785-1-P-A, Revision 1, dated October 1984.

will be revised accordingly to reflect the basis of the proposed method as described in the license amendment request. Accordingly, the NRC staff has included the following wording on the LGS, Units 1 and 2, amendment pages:

> This license amendment is effective as of its date of issuance and shall be implemented within 60 days. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report to reflect the revised main steam line break analysis.

The GEH analysis considered several reactor coolant mass releases, reactor power and flow conditions including conservative operating conditions. The NRC staff reviewed this information and concluded that the approach was reasonable.

The GEH analysis assumes that the initial nuclear system pressure is 1060 pounds per square inch atmospheric and, as energy is removed from the system, the pressure drops. This new assumption of decreasing pressure is more realistic than the current assumption of constant pressure. Therefore, the NRC staff concludes that this change in the MSLB analysis is acceptable.

As described in Section 3.0 of Attachment 1 to the licensee's application dated November 15, 2013, the GEH analysis determined that the most limiting reactor conditions that produce the maximum total coolant mass release, through the MSLB, is the Startup/Hot-Standby condition (4% power and 35% flow). At these conditions, and assuming the MSIVs close within 6 seconds (i.e., instead of the current 5.5 seconds), the total coolant mass release would be 115,700 lbm. This mass release is higher than the current assumed coolant mass release of 108,785 lbm, as shown in UFSAR Table 15.6.4 (i.e., since the current analysis is based on the MSIVs closing in 5.5 seconds). However, the value in the GEH analysis for the proposed amendment is bounded by the 140,000 lbm value assumed in the current MSLB analysis used for the radiological consequence evaluation. As such, the NRC staff concludes that the changes to the MSLB analysis, based on the proposed amendment, are acceptable. The impact on the radiological consequences due to the changes in the MSLB analysis is discussed further in SE Section 3.3 below.

3.2.2 LOCA Analysis

The latest 10 CFR 50.46 annual report for LGS, Units 1 and 2, was submitted by the licensee to the NRC on November 7, 2014 (ADAMS Accession No. ML14311A290). The current LOCA model assessments for each of the units, for GE14 and GNF2 fuel types, determined that the calculated net peak cladding temperatures (PCTs) are as follows:

	Unit 1	**Unit 2**
Net PCT (GE14)	1800 °F	1800 °F
Net PCT (GNF2)	1985 °F	1985 °F

The current calculated PCT values are well within the limit of 2200 °F specified in 10 CFR 50.46(b)(1).

As discussed in Section 3.0 of Attachment 1 to the licensee's application dated November 15, 2013, the current analysis of record for the LGS MSLB LOCA PCT response is documented in NEDC-32170P, Revision 2, dated May 1995. The licensee stated that the MSLB LOCA PCT response is not affected by the proposed amendment as the MSLB LOCA event sequence involves initiation of the automatic depressurization system (ADS), which occurs well after closure of the MSIVs. The MSLB LOCA PCT occurs after ADS initiation. Moreover, an MSLB outside containment is not limiting with regard to PCT. The most limiting case for PCT is the recirculation suction line break. As such, the NRC staff concludes that the proposed change is acceptable with respect to the LOCA analysis.

3.2.3 Reactor Systems Conclusion

Based on the discussion in SE Sections 3.2.1 and 3.2.2, the NRC staff concludes that the proposed change from 0.5 seconds to 1.0 seconds, for the MSIVs to start to close on a high flow signal, is acceptable with respect to the impacts on the accident analysis.

3.3 Radiological Consequences Review

The NRC staff reviewed the proposed amendment from a radiological consequences perspective as described in SE Sections 3.3.1 and 3.3.2 below.

3.3.1 Impact on Radiological Consequences

The NRC staff reviewed the proposed change from 0.5 seconds to 1.0 seconds, for the MSIVs to start to close on a high flow signal, with respect to the impact of the proposed changes on previously-analyzed DBA radiological consequences and the acceptability of the revised analysis results. As discussed in SE Section 2.0, the NRC approved the AST methodology and radiological dose consequence analyses for DBAs for LGS, Units 1 and 2, via License Amendments 185 and 146 on August 23, 2006. (ADAMS Accession No. ML062210214).

As discussed in Section 3.0 of Attachment 1 to the licensee's application dated November 15, 2013, the current analysis of record for the MSLB radiological evaluation is documented in LGS calculation LM-0644, Revision 1, and UFSAR Section 15.6.4. The current sequence of events and approximate time required to respond to an MSLB event are provided in UFSAR Table 15.6-8. The current MSLB radiological consequence analysis results are provided in UFSAR Table 15.6-11.

As discussed in Section 3.0 of Attachment 1 to the licensee's application dated November 15, 2013, the postulated MSLB accident assumes a double-ended break of one MSL outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. However, for the MLSB dose calculations, the break mass released is a bounding maximized value of 140,000 lbm of water that is based on the guidance in SRP 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)," Revision 2.

The licensee stated that two activity release cases, corresponding to the pre-accident spike and maximum equilibrium concentration allowed by the TSs of 4.0 microcuries/gram (μCi/gm) and 0.2 μCi/gm dose equivalent I-131 respectively, were assumed. The inhalation Committed Effective Dose Equivalent dose conversion factors from Federal Guidance Report 11 were used for the normalized Dose Equivalent I-131 determination. The released activity assumptions were based on the guidance provided in Appendix D of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792).

The licensee stated that the analysis assumes an instantaneous ground level release. For the control room dose calculations, the released reactor coolant and steam is assumed to expand to a hemispheric volume at atmospheric pressure and temperature (consistent with an assumption of no turbine building credit). This hemisphere is then assumed to move at a speed of one meter per second downwind past the control room intake. No credit is taken for buoyant rise of the steam cloud or for decay, and dispersion of the activity of the plume was conservatively ignored. For offsite locations, the buoyant rise of the steam cloud is similarly ignored, and the ground level dispersion is based on the conservative and simplified methodology of Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)," dated March 1971.

The current MSLB analysis assumes the MSIVs start to close at 0.5 seconds on high flow signal and are fully closed at 5.5 seconds. As discussed in Section 3.0 of Attachment 1 to the licensee's application dated November 15, 2013, the flow in each line is limited by critical flow at the flow limiter to a maximum of 200 percent of rated flow for each line. Only steam will be released from the broken end of the steam line. Rapid depressurization of the reactor pressure vessel causes the water level to rise, resulting in a steam/water mixture flowing from the break until the MSIVs are closed. In the current MSLB analysis, the total integrated mass leaving the reactor pressure vessel through the steam line break is 108,785 lbm of which 88,333 lbm is liquid and 20,452 lbm is steam.

As discussed above in SE Section 3.3.1, a new MSLB analysis was performed by GEH to evaluate the change in the MSL mass flow resulting from increasing the MSIV closure initiation signal time from 0.5 seconds to 1.0 seconds. The GEH analysis determined there would be a 6,915 lbm increase in the total coolant mass release (from 108,785 lbm to 115,700 lbm).

The MSLB dose calculation (documented in LGS calculation LM-0644) uses the SRP 15.6.4 bounding mass release value of 140,000 lbm, discussed above. This input parameter is not impacted by the increase in the instrument response time as shown by the results of the GEH analysis. The total coolant mass release of 115,700 lbm for the 1.0-second delay case is still well bounded by the 140,000 lbm input. Therefore, the MSLB accident doses for the Control Room, Exclusion Area Boundary and Low Population Zone shown in UFSAR Table 15.6-11 remain unchanged.

3.3.2 Radiological Consequences Conclusion

The NRC staff evaluated the proposed amendment on the impact of the LGS design basis dose consequence analyses to ensure that the modification to the isolation response time will not result in an increase in the dose consequences and that the resulting calculated doses will remain within the design criteria specified in 10 CFR 50.67, and the accident specific design criteria outlined in Regulatory Guide 1.183. Since the MSLB accident doses remain unchanged, the NRC staff concludes that the proposed amendment is acceptable from a dose consequence perspective.

3.4 Technical Evaluation Conclusion

Based on the evaluation in SE Sections 3.1, 3.2, and 3.3, the NRC staff concludes that the proposed amendment is acceptable. The staff further concludes that there is reasonable assurance that the proposed changes will continue to meet the requirements in 10 CFR 50.36, 10 CFR 50.46, 10 CFR 50.67, and GDCs 10, 13, 15, 19, 20, and 29.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (79 FR 6642). Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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R. Ennis

Date: January 7, 2015

January 7, 2015

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

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: MAIN STEAM LINE FLOW-HIGH ISOLATION RESPONSE TIME CHANGE (TAC NOS. MF3085 AND MF3086)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment Nos. 214 and 175 to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station, Units 1 and 2, respectively. These amendments consist of changes to the Technical Specifications (TSs) and Facility Operating Licenses in response to your application dated November 15, 2013, as supplemented by letters dated April 16, 2014, September 11, 2014, and November 7, 2014.

The amendments revise the TS requirements related to the response time for the main steam line flow-high isolation function.

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

Sincerely,
/RA/
Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. Amendment No. 214 to Renewed NPF-39
2. Amendment No. 175 to Renewed NPF-85
3. Safety Evaluation

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ADAMS Accession No: ML14344A681 *** via e-mail**

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