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

November 21, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS TO REDUCE STEAM DOME PRESSURE SPECIFIED IN REACTOR CORE SAFETY LIMITS (CAC NOS. MF7263 AND MF7264)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment Nos. 222 and 183 to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2, respectively. These amendments consist of changes to the technical specifications (TSs) in response to your application dated January 15, 2016, as supplemented by letters dated April 19, 2016; May 9, 2016; and June 21, 2016.

The amendments reduce the reactor vessel steam dome pressure specified in the TSs for the reactor core safety limits. The amendments also revise the setpoint and allowable value for the main steam line low pressure isolation function in the TSs. The changes address a Title 10 of the *Code of Federal Regulations* Part 21 issue concerning the potential to violate the safety limits during a pressure regulator failure maximum demand (open) transient.

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. 222 to Renewed NPF-39
2. Amendment No. 183 to Renewed NPF-85
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
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 January 15, 2016, as supplemented by letters dated April 19, 2016; May 9, 2016; and June 21, 2016, 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) <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. 222, 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 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen S. Koenick, Acting 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: November 21, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 222

LIMERICK GENERATING STATION, UNIT 1

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 a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
2-1	2-1
3/4 3-18	3/4 3-18

(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 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) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 222, 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.

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.10 for two recirculation loop operation and shall not be less than 1.14 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.10 for two recirculation loop operation or less than 1.14 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION			
a.	Reactor Vessel Water Level		
	1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
	2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b.	DELETED	DELETED	DELETED
c.	Main Steam Line Pressure - Low	≥ 840 psig	≥ 821 psig
d.	Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e.	Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f.	Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g.	Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h.	Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION			
a.	Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b.	Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c.	Manual Initiation	N.A.	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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 183
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 January 15, 2016, as supplemented by letters dated April 19, 2016; May 9, 2016; and June 21, 2016, 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.

Enclosure 2

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. 183, 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 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen S. Koenick, Acting 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: November 21, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 183

LIMERICK GENERATING STATION, UNIT 2

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 a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
2-1	2-1
3/4 3-18	3/4 3-18

(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. 183, 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.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 for two recirculation loop operation and shall not be less than 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 for two recirculation loop operation or less than 1.12 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATION CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ 840 psig	≥ 821 psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	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. 222 AND 183

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 January 15, 2016 (Reference 1), as supplemented by letters dated April 19, 2016 (Reference 2); May 9, 2016 (Reference 3); and June 21, 2016 (Reference 4) Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for the Limerick Generating Station (LGS), Units 1 and 2. The proposed amendments would reduce the reactor vessel steam dome pressure associated with the technical specification (TS) safety limits specified in TS 2.1.1, "Thermal Power, Low Pressure or Low Flow," and TS 2.1.2, "Thermal Power, High Pressure and High Flow." The amendments would also revise the setpoint and allowable value for the main steam line low pressure isolation function in TS Table 3.3.2-2, "Isolation Actuation Instrumentation Setpoints." The proposed changes address a Title 10 of the *Code of Federal Regulations* (10 CFR) Part 21 issue concerning the potential to violate the safety limits during a pressure regulator failure maximum demand (open) (PRFO) transient.

The supplements dated April 19, 2016; May 9, 2016; and June 21, 2016, 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 or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 15, 2016 (81 FR 13842).

2.0 REGULATORY EVALUATION

2.1 Background

LGS, Units 1 and 2, TS 2.1.1, currently requires that thermal power not exceed 25 percent of rated thermal power (RTP) when reactor vessel steam dome pressure is less than 785 pounds per square inch gauge (psig) or core flow is less than 10 percent of rated flow. In a letter dated March 29, 2005 (Reference 5), General Electric (GE) submitted a 10 CFR Part 21 notification to the NRC. In the Part 21 notification, GE reported that earlier computer analytical models predicted that during a PRFO transient, a reactor level swell would result in a turbine trip and subsequent reactor scram. However, newer computer models predict that level may not increase to the turbine trip setpoint and may actually be terminated by a main steam isolation valve (MSIV) closure scram at the low pressure isolation setpoint (LPIS). GE concluded that, depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could briefly decrease to below 785 psig while thermal power exceeds 25 percent of RTP, which would be a violation of the TS 2.1.1 safety limit. The GE notification indicated that a number of boiling-water reactor (BWR) plants, including LGS, Units 1 and 2, were affected. Initially, the Boiling Water Reactor Owners Group (BWROG) attempted to resolve the 10 CFR Part 21 issue. On July 18, 2006, the Technical Specifications Task Force (TSTF) and the BWROG submitted Improved Standard Technical Specifications Change (TSTF) Traveler TSTF-495, Revision 0, "Bases Change to Address GE Part 21 SC05-03" (Reference 6) to the NRC for review. The letter stated, in part, that TSTF-495 only affects the TS Bases and would be able to be adopted by plants without requesting a license amendment from the NRC. Specifically, the proposed change would modify the TS Bases to clarify that the reactor core safety limits were not considered to apply to momentary depressurization transients. In a letter to the TSTF dated August 27, 2007 (Reference 7), the NRC staff stated that TSTF-495, Revision 0, could not be approved. The staff's safety evaluation (SE) enclosed with the letter stated, in part:

> The staff agrees with the applicant's position that the PRFO transient does not threaten fuel cladding integrity, since the margin to SLMCPR [safety limit minimum critical power ratio] increases with decreasing reactor pressure. However, the staff is concerned that in some depressurization events which occur at or near full power, there may be enough bundle stored energy to cause some fuel damage. If a reactor scram does not occur automatically, the operator may have insufficient time to recognize the condition and to take the appropriate actions to bring the reactor to a safe configuration.

Based on the above considerations, the NRC staff's SE concluded that TSTF-495, Revision 0, was unacceptable. Consequently, the BWROG discontinued the effort to resolve the issue generically. Several approaches to resolve this issue were considered at periodic BWROG meetings but not adopted, because a generic approach applicable to all BWROG members and fuel vendors could not be identified.

Subsequently, affected BWR licensees have proposed resolution of the Part 21 issue on a plant-specific basis by submittal of LARs that lower the reactor steam dome pressure value in the TS safety limits. This approach takes advantage of the fact that some advanced fuel designs have an NRC-approved critical power correlation with a lower-bound pressure significantly below the reactor steam dome pressure specified in the TS safety limits. With respect to LGS, Units 1 and 2, the licensee proposes to utilize this approach and reduce the reactor steam dome pressure, in TS 2.1.1 and TS 2.1.2, consistent with the approved lower-bound pressure for the critical power correlations for the fuel currently used in the LGS, Units 1 and 2, cores.

2.2 <u>Proposed TS Changes</u>

Consistent with the plant-specific approach discussed above, the licensee proposes to reduce the reactor steam dome pressure, specified in TS 2.1.1 and TS 2.1.2, from 785 psig to 700 pounds per square inch atmospheric (psia)[1]. There would be no other changes to these TSs.

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[1] The application dated January 15, 2016, initially proposed that the steam dome pressure be changed to 685 psig. In response to a request for additional information from the NRC staff, the licensee, in its supplement dated April 19, 2016, proposed that the steam dome pressure be changed to 700 psia (i.e., approximately 685.3 psig).

For LGS, Units 1 and 2, the LAR would revise TS 2.1.1 to read as follows:

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

For LGS Unit 1, the LAR would revise TS 2.1.2 to read as follows:

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.10 for two recirculation loop operation and shall not be less than 1.14 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.10 for two recirculation loop operation or less than 1.14 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

For LGS Unit 2, the LAR would revise TS 2.1.2 to read as follows:

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 for two recirculation loop operation and shall not be less than 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 for two recirculation loop operation or less than 1.12 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

As discussed in the licensee's application dated January 15, 2016, in response to the Part 21 issue, the BWROG commissioned development of a methodology for plants to assess the adequacy of their MSIV closure at the LPIS and to provide a set of recommendations for what actions should be taken based on the outcome of their assessment. Based on the results of the BWROG's efforts, the licensee determined that the current MSIV LPIS analytical limit[2] of 720 psig, for LGS, Units 1 and 2, is not sufficient to preclude reactor vessel steam dome pressure from falling below the value of 700 psia (specified in proposed TS 2.1.1 and TS 2.1.2), while above 25 percent power for operation during a PRFO event. As a result, a change to the MSIV LPIS analytical limit from 720 psig to 805 psig was determined to be needed. Based on this proposed change to the analytical limit, the licensee determined that changes to the MSIV LPIS allowable value and trip setpoint specified in the LGS TSs would be required.

Accordingly, in the application dated January 15, 2016, licensee proposed the following changes to the main steam isolation function in TS Table 3.3.2-2:

1) Increase the trip setpoint for Function 1.c, "Main Steam Line Pressure – Low," from ≥ 756 psig to ≥ 840 psig.

2) Increase the allowable value for Function 1.c, "Main Steam Line Pressure – Low," from ≥ 736 psig to ≥ 821 psig.

2.3 Regulatory Requirements and Guidance

The regulatory requirements and guidance that the NRC staff considered in its review of this LAR are described below.

General Design Criteria

Section 3.1 of the LGS Updated Final Safety Analysis Report (UFSAR) discusses the extent to which the design of LGS conforms to the General Design Criteria specified in Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants."

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 (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). Section 3.1 of the LGS UFSAR states that the reactor core and associated coolant, control, and protection systems are designed to meet the requirements of GDC-10.

[2]The analytical limit is a calculated variable established by the safety analysis to ensure that the associated safety limit is not exceeded. The analytical limit is used in a setpoint calculation to determine the setpoint and allowable value.

Technical Specification Requirements

In 10 CFR 50.36, "Technical specifications," the NRC established requirements related to the content of TSs. Pursuant to 10 CFR 50.36(c), TSs must include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (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)(1), safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If a safety limit is exceeded, the reactor must be shut down. The safety limits for LGS, Units 1 and 2, are specified in TS 2.1.

As discussed in Section 4.1 of Attachment 1 of the licensee's application dated January 15, 2016 (Reference 1), the fuel cladding is one of the physical barriers that separates the radioactive materials from the environment. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Fuel cladding perforations can result from thermal stresses, which can occur from reactor operation significantly above design conditions. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling (OTB) have been used to mark the beginning of the region in which fuel cladding damage could occur. The reactor core safety limits are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur, due to OTB, if the safety limits are not exceeded.

Guidance Documents

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereinafter referred to as "SRP") (Reference 8), provides guidance on, among other things, the acceptability of the reactivity control systems, the reactor core, and fuel system design. Relevant sections of the SRP used in review of this LAR include the following:

- Chapter 4, Section 4.2, "Fuel System Design," Revision 3, dated March 2007 (Reference 9). Section 4.2 specifies the criteria for evaluation of fuel damage and whether fuel designs meet the SAFDLs.

- Chapter 4, Section 4.4, "Thermal and Hydraulic Design," Revision 2, dated March 2007 (Reference 10). Section 4.4 provides guidance on the review of thermal-hydraulic design in meeting the requirements of GDC-10 and the fuel design criteria established in SRP Section 4.2. It states that the critical power ratio (CPR) is to be established such that at least 99.9 percent of fuel rods in the core would not be expected to experience departure from nucleate boiling or OTB during normal operation or AOOs.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of Changes to TSs 2.1.1 and 2.1.2

Background

Each fuel vendor has developed critical power correlations valid over specified pressure and flow ranges (mass flow rates) that are approved by the NRC. These critical power correlations have become increasingly fuel-design dependent as advanced fuel designs have evolved. The critical power correlations for some advanced fuel designs have received NRC approval down to a lower pressure than those approved previously. If justified, the lower bound of the extended pressure ranges for these advanced fuel designs can be used to establish a lower reactor vessel steam dome pressure than specified in the TSs for previous fuel designs. As such, a wider pressure range would be available for a PRFO transient to demonstrate compliance with MCPR limits. As discussed above in SE Sections 2.1 and 2.2, the licensee proposes to reduce the steam dome pressure specified in TS 2.1.1 and TS 2.1.2 from 785 psig to 700 psia based on the lower-bound pressure for the critical power correlation for the fuel currently used in the LGS, Units 1 and 2, cores. As discussed in the licensee's application dated January 15, 2016, LGS, Unit 1, currently has a mixed core of Global Nuclear Fuel (GNF) GNF2 fuel and GE14 fuel. The LGS, Unit 2, reactor currently contains GNF2 fuel only.

The OTB in BWR fuel assemblies, during both steady-state and reactor transient conditions, can be predicted by the GE critical quality-boiling length correlation, better known as the GEXL correlation. As discussed in the References 11 and 12, in the core design process, the GEXL correlation is used to determine the expected thermal margin for the operating cycle. In the safety analysis process, the GEXL correlation is used in the determination of the change in CPR during postulated transients and in the determination of an acceptable MCPR safety limit.

The critical power correlation (i.e., GEXL correlation) for GNF2 fuel is referred to as the GEXL17 correlation. The critical power correlation for GE14 fuel is referred to as the GEXL14 correlation.

GNF Licensing Topical Report (LTR) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (Reference 13), provides generic information relative to the fuel design and analyses of BWRs that use the GE and GNF fuel designs. This LTR (referred to as GESTAR II) consists of a description of the fuel licensing criteria and fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases. LGS TS 6.9.1.9 through TS 6.9.1.12 describe the administrative controls requirements for the LGS Core Operating Limits Report. In accordance with TS 6.9.1.10, the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the latest approved version of GESTAR II.

NRC Approval of GEXL17 Correlation

The GEXL17 correlation is documented in GNF report NEDC-33292P, Revision 3, "GEXL17 Correlation for GNF2 Fuel," dated June 2009 (Reference 11).

In a letter dated March 5, 2010 (Reference 14), GNF submitted proposed Amendment No. 33 to GESTAR II for NRC review and approval. The letter also provided GNF report NEDC-33270P, Revision 3, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," dated March 2010. The GNF report documented the completion of the requirements for the new GNF2 fuel design per the criteria in GESTAR II. Section 3.8.3 of GNF report NEDC-33270P discusses the GESTAR II criteria for critical power correlations and references GNF report NEDC-33292P as the basis for the GEXL17 correlation for the GNF2 fuel. In a letter dated August 30, 2010 (Reference 15), the NRC staff approved Amendment No. 33 to GESTAR II.

Based on the above, the GEXL17 correlation, as documented in GNF report NEDC-33292P, is approved for use by the NRC per the approval of Amendment 33 to GESTAR II. Amendment No. 33 was incorporated in Revision 17 to GESTAR II by GNF letter dated September 22, 2010 (Reference 16).

NRC Approval of GEXL14 Correlation

The GEXL14 correlation is documented in GNF report NEDC-32851P-A, Revision 5, "GEXL14 Correlation for GE14 Fuel," dated April 2011 (Reference 12). The NRC's approval of this topical report is documented in the staff's SE dated April 3, 2007, included in the report.

Lower Bound Pressure Limit for GEXL17 and GEXL14 Correlations

Section 3.8.3 of GNF report NEDC-33270P (Reference 14) includes the pressure range over which the GEXL17 correlation is valid for GNF2 fuel, consistent with the information provided in Table 5-4 of GNF2 report NEDC-33292P (Reference 11). As discussed in Section 3.0 of Attachment 1 to the licensee's application dated January 15, 2016, the lower bound pressure limit for the GEXL17 correlation is 700 psia. Similarly, Section 5.2 of GNF report NEDC-32851P-A (Reference 12) includes the pressure range over which the GEXL14 correlation is valid for GE14 fuel. Section 5.2 of the report states that the lower bound pressure limit for the GEXL14 correlation is 700 psia.

Conclusions Regarding Changes to TS 2.1.1 and TS 2.1.2

Based on the above, the NRC staff finds that:

1) The use of the GEXL17 correlation for GNF2 fuel is considered an NRC-approved method, consistent with the Revision 17 of GESTAR II.

2) The use of the GEXL14 correlation for GE14 fuel is considered an NRC-approved method, consistent with NRC-approved GNF report NEDC-32851P-A.

3) The use of GESTAR II for development of the LGS, Units 1 and 2, core operating limits is required by TS 6.9.1.10.

4) The use of the GEXL17 and GEXL14 correlations will ensure that valid CPR calculations are performed for the AOOs applicable to LGS, Units 1 and 2, including the PRFO transient.

5) The proposed 700 psia reactor steam dome pressure in TS 2.1.1 and TS 2.1.2 is justified based on the lower bound pressure associated with the GEXL17 correlation for GNF2 fuel and the GEXL14 correlation for GE14 fuel.

Based on the above findings, the NRC staff concludes that as long as the core pressure and flow are within the range of validity of the GEXL17 and GEXL14 correlations, the proposed reactor steam dome pressure changes to the safety limits in TS 2.1.1 and TS 2.1.2 provide reasonable assurance that 99.9 percent of the fuel rods in the core are not expected to experience OTB during normal operation or AOOs. This will continue to ensure that SAFDLs are not exceeded during normal operation or AOOs, consistent with the requirements in GDC-10. Furthermore, the NRC staff concludes that the proposed changes establish reactor core safety limits to protect the integrity of the fuel cladding barrier and guard against an uncontrolled release of radioactivity, consistent with the requirements in 10 CFR 50.36(c)(1). Based on the above conclusions, the NRC staff further concludes that the proposed changes to TS 2.1.1 and TS 2.1.2 are acceptable.

The NRC staff notes that if LGS, Units 1 and 2, transition to a new fuel design, the licensee should review the critical power correlation to determine if further changes to the reactor core safety limits are required. As long as the lower bound pressure associated with the correlation for the new fuel design is less than or equal to the TS 2.1.1 and TS 2.1.2 reactor steam dome pressure, then an LAR would not be required. However, if the lower bound pressure associated with the critical power correlation for the new fuel design is higher than the reactor steam dome pressure specified in TS 2.1.1 and TS 2.1.2, an LAR would be required.

3.2 Evaluation of Changes to TS Table 3.3.2-2

As discussed above in SE Section 2.2, the licensee determined that the current MSIV LPIS analytical limit of 720 psig is not sufficient to preclude reactor vessel steam dome pressure from falling below the value of 700 psia (specified in proposed TS 2.1.1 and TS 2.1.2) while above 25 percent power for operation during a PRFO event. As a result, a change to the MSIV LPIS analytical limit from 720 psig to 805 psig was determined to be needed. Based on this proposed change to the analytical limit, the licensee determined that the MSIV LPIS allowable value and trip setpoint specified in TS Table 3.3.2-2 would need to be revised as follows:

1) Increase the trip setpoint for Function 1.c, "Main Steam Line Pressure – Low," from ≥ 756 psig to ≥ 840 psig.

2) Increase the allowable value for Function 1.c, "Main Steam Line Pressure – Low," from ≥ 736 psig to ≥ 821 psig.

To assess the acceptability of the TS Table 3.3.2-2 changes, the NRC staff requested that the licensee submit the revised instrument setpoint/loop uncertainty (LU) calculation for staff review. In Attachment 4 to its supplement dated April 19, 2016, the licensee provided a partial copy of calculation LI-00032, "LU Calculation for PT-001-2N076C." A complete version of the calculation was provided in the attachment to the licensee's supplement dated June 21, 2016.

The LGS, Units 1 and 2, setpoint methodology is contained in Exelon Procedure CC-MA-103-2001. This procedure is based on the NRC-approved GE Topical Report NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996. The NRC staff found the setpoint methodology acceptable as discussed in an NRC letter dated February 16, 1995, "Revised Maximum Authorized Thermal Power Limit, Limerick Generating Station, Unit No. 2" (Reference 17).

The objective of calculation LI-00032 was to determine the nominal trip setpoint, actual trip setpoint, and the allowable value for the instrumentation associated with TS Table 3.3.2-2, Function 1.c. This calculation analyzed LGS, Unit 2, instrumentation loop PT-001-2N076C as a bounding case for the other three Unit 2 instrumentation loops and four Unit 1 instrumentation loops providing the same function. The licensee determined this loop was bounding based on review of factors such as instrument manufacturer and model number and instrument location environmental parameters.

The process parameters used in the calculation remain unchanged. This calculation was performed utilizing normal environmental conditions (reference Section 2.2.3 of the calculation) based on the design information contained in Section 15.1.3 of the LGS UFSAR. UFSAR Section 15.1.3 indicates that the design-basis event for the isolation of the main steam line as a result of low steam line pressure is a failure of the main turbine pressure regulator. This failure will not result in release of steam to the turbine enclosure environment. Therefore, the pressure transmitters will not be subjected to any harsh environment effects when accomplishing their intended safety function.

The calculation methodology has not changed. However, due to changes in the safety limit and the associated changes in the analytical limit, it was necessary for the licensee to recalculate a new trip setpoint and the new allowable value based on the new trip setpoint. Since the loop instruments and their uncertainties remain unchanged, the total loop uncertainty has not changed, and it remains as ± 21.47 psig (as shown in Section 7.4 of the calculation). Based on the new analytical limit value of 805 psig, the recalculated nominal trip setpoint is 822.66 psig as shown in Section 7.5 of the calculation. As shown in Section 7.6 of the calculation, additional margin of 17.345 psig was added to the nominal trip setpoint. As a result, the actual trip setpoint was selected to be 840 psig. In addition, as a result of the change in analytical limit from 720 psig to 805 psig, the allowable value changed from 736 psig to 821 psig (as shown in Section 7.7 of the calculation).

Based on review of the calculation, the NRC staff finds that:

1) The licensee used an NRC-approved methodology to determine the revised actual trip setpoint and allowable value.

2) The results of the calculation are consistent with the trip setpoint and allowable value for the proposed changes to TS Table 3.3.2-2.

3) The proposed changes to TS Table 3.3.2-2 provide reasonable assurance that the reactor vessel steam dome pressure will not fall below the value of 700 psia (specified in proposed TS 2.1.1 and TS 2.1.2) while above 25 percent power for operation during a PRFO event. As such, this resolves the 10 CFR Part 21 issue discussed above in SE Section 2.1.

Based on the above findings, the NRC staff concludes that the proposed changes to TS Table 3.3.2-2 are acceptable.

3.3 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 and 3.2, the NRC staff concludes that the proposed amendments are acceptable.

The licensee's supplement dated April 19, 2016, provided revised TS Bases pages to be implemented with the associated TS changes. These pages were provided for information only and will be revised by the licensee in accordance with the TS Bases Control Program described in LGS TS 6.8.4.h, "Technical Specifications (TS) Bases Control Program."

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the 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 (81 FR 13842). 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.

7.0 REFERENCES

1. Exelon letter to the NRC, "License Amendment Request – Proposed Revision to Technical Specifications in Response to GE Energy – Nuclear 10 CFR Part 21 Safety Communication SC05-03," dated January 15, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16015A316).

2. Exelon letter to the NRC, "Response to Draft Request for Additional Information Regarding Proposed Revision to Technical Specifications in Response to GE Energy – Nuclear 10 CFR Part 21 Safety Communication SC05-03," dated April 19, 2016 (ADAMS Accession No. ML16110A392).

3. Exelon letter to the NRC, “License Amendment Request – Supplement, Proposed Revision to Technical Specifications in Response to GE Energy – Nuclear 10 CFR Part 21 Safety Communication SC05-03,” dated May 9, 2016 (ADAMS Accession No. ML16131A698). Note, this letter includes a proprietary non-public attachment (ADAMS Accession No. ML16131A699).

4. Exelon letter to the NRC, “License Amendment Request – Supplement, Proposed Revision to Technical Specifications in Response to GE Energy – Nuclear 10 CFR Part 21 Safety Communication SC05-03,” dated June 21, 2016 (ADAMS Accession No. ML16173A395).

5. GE letter to the NRC, “10 CFR 21 Reportable Condition Notification: Involving Potential to Exceed Low Pressure Technical Specification Safety Limit,” dated March 29, 2005 (ADAMS Accession No. ML050950428).

6. TSTF letter to the NRC transmitting TSTF-495, Revision 0, “Bases Change to Address GE Part 21 SC05-03,” dated July 18, 2006 (ADAMS Accession No. ML061990227).

7. NRC letter to Technical Specification Task Force, “Denial of TSTF-495, Revision 0, ‘Bases Change to Address GE Part 21 SC05-03’,” dated August 27, 2007 (ADAMS Accession No. ML072340113).

8. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (hereinafter referred to as “SRP”) (ADAMS Accession No. ML070810350)

9. NUREG-0800, Chapter 4, Section 4.2, “Fuel System Design,” Revision 3, dated March 2007 (ADAMS Accession No. ML070740002).

10. NUREG-0800, Chapter 4, Section 4.4, “Thermal and Hydraulic Design,” Revision 2, dated March 2007 (ADAMS Accession No. ML070550060).

11. Global Nuclear Fuel (GNF) report NEDC-33292P, “GEXL17 Correlation for GNF2 Fuel,” Revision 3, dated June 2009 (ADAMS Accession No. ML091830641 (non-publicly available)).[3]

12. GNF letter MFN 11-140 to the NRC, “Acceptance Version of Global Nuclear Fuel (GNF) Topical Report (TR) NEDC-32851P, Revision 5, GEXL14 Correlation for GE14 Fuel,” dated May 6, 2011 (ADAMS Package Accession No. ML111290540).

13. GNF LTR NEDE 24011P-A, “General Electric Standard Application for Reactor Fuel” (latest approved revision).

14. GNF letter MFN 10-045 to the NRC, “Amendment 33 to NEDE-24011-P, General Electric Standard Application for Reactor Fuel (GESTAR II) and GNF2 Advantage

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[3]This proprietary report was submitted to the NRC as Enclosure 4 to GNF letter MFN 09-436 dated June 30, 2009 (ADAMS Accession No. ML091830614). Enclosure 5 to the letter (ADAMS Accession No. ML091830624) is a publicly available version of the report.

Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDC-33270P, Revision 3, March 2010," dated March 5, 2010 (ADAMS Package Accession No. ML100700464).

15. NRC letter to GNF, "Final Safety Evaluation for Amendment 33 to Global Nuclear Fuel Topical Report NEDE-24011-P, 'General Electric Standard Application for Reactor Fuel (GESTAR II)'," dated August 30, 2010 (ADAMS Accession No. ML102280144).

16. GNF letter MFN 10-250 to the NRC, "Accepted Proprietary and Non-Proprietary Versions of Revision 17 to NEDE-24011-P, General Electric Standard Application for Reactor Fuel (GESTAR II), Main and United States Supplement," dated September 22, 2010 (ADAMS Package Accession No. ML102660094).

17. NRC letter to PECO Energy Company, "Revised Maximum Authorized Thermal Power Limit, Limerick Generating Station, Unit No. 2," dated February 16, 1995 (ADAMS Accession No. ML011560773).

Principal Contributors: R. Ennis
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M. Hardgrove

Dated: November 21, 2016

November 21, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS TO REDUCE STEAM DOME PRESSURE SPECIFIED IN REACTOR CORE SAFETY LIMITS (CAC NOS. MF7263 AND MF7264)

Dear Mr. Hanson:

The Commission has issued the enclosed Amendment Nos. 222 and 183 to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2, respectively. These amendments consist of changes to the technical specifications (TSs) in response to your application dated January 15, 2016, as supplemented by letters dated April 19, 2016; May 9, 2016; and June 21, 2016.

The amendments reduce the reactor vessel steam dome pressure specified in the TSs for the reactor core safety limits. The amendments also revise the setpoint and allowable value for the main steam line low pressure isolation function in the TSs. The changes address a Title 10 of the *Code of Federal Regulations* Part 21 issue concerning the potential to violate the safety limits during a pressure regulator failure maximum demand (open) transient.

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. 222 to Renewed NPF-39
2. Amendment No. 183 to Renewed NPF-85
3. Safety Evaluation

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DATE	11/02/2016	10/05/2016	10/13/2016	10/13/2016	10/14/2016
OFFICE	OGC	DORL/LPL1-2/BC(A)	DORL/LPL1-2/PM		
NAME	MYoung	SKoenick	REnnis		
DATE	11/08/2016	11/21/2016	11/21/2016		

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