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

September 28, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Nuclear Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 − ISSUANCE OF AMENDMENT NOS. 253 AND 215 RE: TECHNICAL SPECIFICATION CHANGES RELATED TO RELOCATION OF PRESSURE AND TEMPERATURE LIMIT CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2020-LLA-0221)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 253 and 215 to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2 (Limerick), respectively, in response to your application dated September 29, 2020.

Pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.90 (10 CFR 50.90), the licensee proposed to revise the Limerick Units 1 and 2 TSs to relocate the pressure-temperature (P-T) limits for the reactor pressure vessel (RPV) to a licensee-controlled PTLR. Specifically, the licensee proposed to (1) modify TS 1.1, “Definitions,” (2) delete the P-T curves in TS 3/4.4.6, (3) add a new Section 6.9.1.13, “REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR), in TS 6.0 to include the reference to the unit-specific PTLRs, and (4) include the P-T limit curves for 57 effective full power years (EFPY) in the PTLR.

The amendments revise the Technical Specification (TS) Section 1.0, “Definitions,” Section 3/4.4.6, “Pressure/Temperature Limits,” and TS Section 6.0, “Administrative Controls,” by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature limit curves with references to the PTLR.

A copy of the related Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

V. Sreenivas, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. Amendment No. 253 to Renewed NPF-39
2. Amendment No. 215 to Renewed NPF-85
3. Safety Evaluation

cc: 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. 253
Renewed License No. NPF-39

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated September 29, 2020, 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. 253, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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Date: 2021.09.28 16:03:42 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 28, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 253

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 attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3	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 a marginal line indicating the areas of change.

Remove Page	Insert Page
1-5	1-5
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
6-18a	6-18a

(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 percent 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. 253, 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.

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.28a The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.13.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the solidification or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With solidification or dewatering, the PCP shall identify the process parameters influencing solidification or dewatering, based on laboratory scale and full scale testing or experience.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limits specified in the PTLR, with:

a. A maximum heatup rate within the limits specified in the PTLR,

b. A maximum cooldown rate within the limits specified in the PTLR,

c. A maximum temperature change within the limits specified in the PTLR during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature within the limits specified in the PTLR when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limits specified in the PTLR as applicable, in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limits specified in the PTLR within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and in accordance with the Surveillance Frequency Control Program during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be within the limits specified in the PTLR:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

    1. ≤ 100°F, in accordance with the Surveillance Frequency Control Program.

    2. ≤ 90°F, in accordance with the Surveillance Frequency Control Program.

b. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during tensioning of the reactor vessel head bolting studs.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factors for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification 3.1.4.3,

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),*

b. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, " August 1996.

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

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

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.13 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

a. Limiting Condition for Operation Section 3.4.6, "RCS Pressure/Temperature Limits"
b. Surveillance Requirement Section 4.4.6, "RCS Pressure/Temperature Limits"

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. BWROG-TP-11-022-A, Revision 1 (SIR-05-044), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

* For Cycle 8, specific documents were approved in the Safety Evaluation dated (5/4/98) to support License Amendment No. (127).

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 215
Renewed License No. NPF-85

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated September 29, 2020, 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. 215, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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Date: 2021.09.28 16:04:21 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 28, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 215

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 attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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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 a marginal line indicating the areas of change.

Remove Page	Insert Page
1-5	1-5
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
6-18a	6-18a

(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. 215, 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.

OPERATIONAL CONDITION - CONDITION
1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS
1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE
1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)
1.28a The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.13.

PRIMARY CONTAINMENT INTEGRITY
1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM
1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With SOLIDIFICATION or dewatering, the PCP shall identify the process parameters influencing SOLIDIFICATION or dewatering based on laboratory scale and full scale testing or experience.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limits specified in the PTLR, with:

a. A maximum heatup rate within the limits specified in the PTLR,

b. A maximum cooldown rate within the limits specified in the PTLR,

c. A maximum temperature change within the limits specified in the PTLR during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature within the limits specified in the PTLR when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limits specified in the PTLR as applicable, in accordance with the Surveillance Frequency Control Program.

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limits specified in the PTLR within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and in accordance with the Surveillance Frequency Control Program during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be within the limits specified in the PTLR:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. ≤ 100°F, in accordance with the Surveillance Frequency Control Program.

   2. ≤ 90°F, in accordance with the Surveillance Frequency Control Program.

b. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during tensioning of the reactor vessel head bolting studs.

INFORMATION CONTAINED ON THIS PAGE HAS BEEN DELETED

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factor for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification 3.1.4.3.

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),

b. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, " August 1996.

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

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

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.13 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

a. Limiting Condition for Operation Section 3.4.6, "RCS Pressure/Temperature Limits"
b. Surveillance Requirement Section 4.4.6, "RCS Pressure/Temperature Limits"

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. BWROG-TP-11-022-A, Revision 1 (SIR-05-044), “Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013.

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 253

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-39

AND AMENDMENT NO. 215

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-85

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated September 29, 2020, (Agencywide Documents and Access Management System (ADAMS) Accession No. ML20273A215), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) to modify Technical Specification (TS) with references to the pressure and temperature limits report (PTLR) at Limerick Generating Station (Limerick), Units 1 and 2.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.90, the licensee proposed to revise the Limerick Units 1 and 2 TSs to relocate the pressure-temperature (P-T) limits for the reactor pressure vessel (RPV) to a licensee-controlled PTLR. Specifically, the licensee proposed to (1) modify TS 1.1, "Definitions," (2) delete the P-T curves in TS 3/4.4.6, (3) add a new Section 6.9.1.13, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), in TS 6.0 to include the reference to the unit-specific PTLRs, and (4) include the P-T limit curves for 57 effective full power years (EFPY) in the PTLR.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.36, "Technical specifications," Paragraph (a), requires that each operating license application for a production or utilization facility include proposed TSs and a summary statement of the bases for such specifications. Paragraph (c) of 10 CFR 50.36 requires, in part, that TSs include the following categories related to facility operation: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

The regulation, 10 CR 50.36(c)(2), "Limiting conditions for operation," states, "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear

reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

The regulations in 10 CFR 50.36(c)(3), "Surveillance requirements," states, "Surveillance requirements are requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

The regulations in 10 CFR 50.36(c)(5), "Administrative controls," states, "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

The regulations in 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation," requires that all operating light-water nuclear power reactors meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

The regulations in 10 CFR Part 50, Appendix G, requires: (1) sufficient fracture toughness for RPV ferritic materials to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests; (2) P-T limits that satisfy the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G, and the minimum temperature requirements during normal heatup, cooldown, and pressure test operations; and (3) applicable surveillance data from RPV material surveillance programs developed in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," be incorporated into the calculations of P-T limits.

The U.S. Nuclear Regulatory Commission (NRC, the Commission) Generic Letter (GL) 96-03, "Relocation of Pressure and Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996 (ADAMS Accession No. ML03111004) permits relocation of the P-T limits from the TS to a PTLR. GL 96-03 recommends that licensees who seek a license amendment for relocation (1) generate their P-T limits in accordance with an NRC-approved methodology, (2) comply with 10 CFR Part 50, Appendices G and H, (3) reference NRC-approved methodologies in the TS, (4) define the PTLR in TSs Section 1.0, (5) develop a PTLR to contain the P-T limit curves, and (6) modify applicable sections of the TS accordingly.

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988 (ADAMS Accession No. ML003740284), describes procedures for calculating the adjusted nil-ductility transition reference temperature RTNDT (ART) due to neutron irradiation on RPVs.

Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," dated March 6, 1992 (ADAMS Accession No. ML031070438) requires that licensees submit their plant specific RPV data to the NRC staff for review. GL 92-01, Revision 1, Supplement 1, dated May 19, 1995 (ADAMS Accession No. ML031070449), requires that licensees provide and assess data from other licensees that could affect their RPV integrity.

The NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," dated October 14, 2014 (ADAMS Accession No. ML14149A165) provides

evaluation guidance for P-T limit curves and PTLRs, including the consideration of neutron fluence and structural discontinuities in the development of P-T limit curves (ADAMS Accession No. ML14149A165).

The NRC guidance, in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Section 5.3.2, Revision 2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock” (ADAMS Accession No. ML070380185), provides an acceptable method for determining the P-T limits based on the methodology of ASME Code, Section XI, Appendix G.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001 (ADAMS Accession No. ML010890301), provides guidance regarding neutron fluence calculations.

By letter dated March 21, 2002 (ADAMS Accession No. ML020800488), the NRC approved the use of Technical Specifications Task Force (TSTF) Traveler TSTF-419-A, Revision 0, “Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR,” (ADAMS Accession No. ML012690234). By letter dated August 4, 2011 (ADAMS Accession No. ML110660285), the NRC staff clarified the use of TSTF-419-A.

By letter dated May 16, 2013 (ADAMS Accession No. ML13107A062), the NRC staff approved the use of Boiling Water Reactor (BWR) Owners Group (BWROG) Topical Report BWROG-TP-11-022, Revision 1, to construct P-T limit curves and to relocate the P-T limit curves to a PTLR using TSTF 419-A. By letter dated September 4, 2013 (ADAMS Accession No. ML13277A557), the BWROG issued the final NRC-approved version of BWROG-TP-11-022-A, Revision 1.

The NRC staff approved the use of BWROG-TP-11-023-A, Revision 0, “Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations,” which provides guidance on analyzing RPV nozzles for the P-T limit curves on June 28, 2013 (ADAMS Accession No. ML13183A017).

3.0 TECHNICAL EVALUATION

3.1 Licensee Proposed Changes to Technical Specifications

The licensee proposed revisions to applicable sections of the TSs are shown in Attachments 1 and 2 to the submittal dated September 29, 2020, as shown below. The licensee stated that the proposed changes are consistent with the guidance in GL 96-03, as supplemented by TSTF-419-A.

(a) TS Section 1.0, “Definitions”

Add a new definition, “Pressure and Temperature Limits Report.” The licensee stated that wording for this definition is consistent with that in TSTF-419-A.

(b) TS Section 3/4.4.6, “Pressure/Temperature Limits”

Delete the P-T curves and the associated TS wording and replace them with references to the PTLR.

(c) TS Section 6.0, “Administrative Controls”

Add a new Section 6.9.1.13 “REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)”. The licensee stated that the format and content are consistent with that in TSTF-419-A. This new section would: (1) identify the individual TS that addresses reactor coolant system P-T limits; (2) reference the NRC-approved topical report that documents PTLR methodologies in a complete citation; and (3) require that the PTLR and any revision or supplement thereto be submitted to the NRC.

3.2 Licensee’s Neutron Fluence Calculations

The proposed P-T limits are based on the application of the BWROG report methodology, an approved generic methodology for generating P-T limits based on the plant-specific adjusted reference temperatures (ARTs), which is consistent with the NRC PTLR development guidance in GL 96-03. Implementation of the BWROG report methodology is documented in the proposed Limerick PTLR.

In March 2001, the NRC staff issued RG 1.190 which provides methods for determining RPV fluence. Fluence calculations are acceptable if they are performed with approved methodologies or with methods which are shown to conform with the guidance in RG 1.190. The licensee submitted fluence calculations for 57 EFPY. These calculations were performed using the Radiation Analysis Modeling Application (RAMA) methodology, which was approved by the NRC staff.

3.3 Licensee Proposed P-T Limits

The proposed Limerick Units 1 and 2 PTLRs are presented in Attachments 11 and 12 to the licensee’s September 29, 2020, letter, respectively. The licensee stated that it prepared the P-T limit curves and PTLR in accordance with GL 96-03; TSTF-419-A; ASME Code, Section XI, Appendix G; BWROG-TP-11-022-A, Revision 1; and BWROG-TP-11-023-A. Specifically, the licensee calculated (1) the P-T limits in accordance with BWROG-TP-11-022-A, Revision 1, BWROG-TP-11-023-A, and the ASME Code, Section XI, Appendix G; (2) neutron fluence in accordance with RG 1.190 using the RAMA computer code, and (3) adjusted nil-ductility transition reference temperature (ART) values for the limiting beltline materials in accordance with RG 1.99, Revision 2.

The licensee provided supporting data and calculations from BWRVIP-135, “Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program Data Source Book and Plant Evaluations,” for determining the proprietary Integrated Surveillance Program (ISP) material chemistry factor values listed in the PTLRs as discussed in Attachment 13 to the September 29, 2020, submittal.

The licensee constructed P-T curves for three plant operating conditions (i.e., the hydrostatic pressure and leak test condition (Curve A), normal operation with core not critical condition (Curve B), and normal operation with core critical condition (Curve C)). In each of Curves A, B, and C, the licensee also constructed a P-T curve for each of three RPV regions---the beltline region, non-beltline region, and bottom head region. The licensee stated that its composite P-T curves contain the bounding P-T curves from the three regions of the RPV, including the shell plates and structural discontinuities such as nozzles.

The licensee used finite element models to generate the thermal and pressure stress distributions for feedwater nozzles, instrumentation nozzle, and low pressure coolant injection (LPCI) nozzle. The licensee used the thermal and pressure stress distributions to derive the thermal stress intensity factor, K_{IT}, and pressure stress intensity factor, K_{ip}. Finally, the licensee constructed P-T curves by comparing K_{IT} and K_{ip} to the material toughness of the RPV materials.

4.0 NRC Staff Evaluation

The NRC staff evaluated (1) neutron fluence calculations in accordance with RG 1.190; (2) the proposed PTLR implementation in accordance with the guidance in GL 96-03 and TSTF-419-A; and (3) the proposed P-T limit curves in accordance with BWROG-TP-11-022-A, Revision 1; BWROG-TP-11-023-A; SRP 5.3.2; 10 CFR Part 50, Appendices G and H; and ASME Code, Section XI, Appendix G.

4.1 Neutron Fluence Evaluation

As noted in Section 3.2 of this safety evaluation (SE), the licensee demonstrated the acceptability of its PTLR by evaluating the PTLR methodology against the seven methodological criteria in Attachment 1 to GL 96-03 for P-T limits relocation. A specific neutron fluence methodology is not included in the BWROG report methodology, therefore, the NRC staff reviewed each of the three topics identified for Criterion 1, which include: to describe how the neutron fluence is calculated; to describe the transport calculation methods, including computer code and formula used to calculate neutron fluence; and to provide the neutron fluence values that are used in the ART calculation. Table 101 of the BWROG report states, "fluence methods and results must comply with RG 1.190 and have NRC approval for use with this [licensing topical report]."

The licensee submitted updated fluence calculations for Limerick, Units 1 and 2, in its the LAR. The licensee updated the fluence calculations using an NRC-approved methodology in accordance with RG 1.190. Fluence values are appropriately projected to 57 EFPY based on using actual operational data for all completed cycles. Cycle-specific fuel designs are used, and cycle-specific EFPY is tracked. Current fuel design is used for the fluence projection.

The NRC staff reviewed the submitted neutron fluence calculations and found that the calculations are representative of past operating conditions. The licensee has shown that the current and future operating conditions would be appropriately accounted for and would result in updated fluence projections when necessary, and the use of the fluence calculational method would be expected to continue to produce best-estimate fluence values within the 20 percent allowance for uncertainty at the 1-sigma level recommended in RG 1.190. This would ensure that the margins provided for fluence in the temperature shift calculations required by 10 CFR Part 50, Appendix G would be bounding of the uncertainties associated with the calculated, best-estimate fluence values.

Based on its review, the NRC staff concludes that the information provided assures that the proposed PTLR and subsequent updates will use appropriate fluence calculational method inputs with a fluence calculational methodology that adheres to RG 1.190. In addition, the NRC staff finds that the use of the methodology and appropriate projections meets the regulatory requirements in 10 CFR 50, Appendix G . Therefore, the NRC staff finds that the fluence calculational method described in Attachments 14 and 15 of the LAR is acceptable for use with

the PTLR methodology based on appropriate 57 EFPY fluence projections and conforms to the guidance provided in RG 1.190.

4.2 PTLR Implementation

The NRC staff evaluated the proposed Limerick, Units 1 and 2 PTLRs in accordance with the seven criteria in Attachment 1 to GL 96-03 as discussed below.

Criterion 1 requires that the PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluence values.

As stated in the LAR, the licensee utilized the BWROG report methodology to generate the P-T limits. The BWROG report methodology is an NRC-approved method for use in generating PTLRs. The PTLR methodology describes the transport calculation methods, including computer codes and formula used to calculate neutron fluences. Section 3 of the PTLR describes the methods and computer codes used.

The NRC staff finds that Limerick, Units 1 and 2 PTLRs have satisfied Criterion 1 of Attachment 1 to GL 96-03 because both PTLRs provide appropriate transportation calculation methods that satisfy RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.”

Criterion 2 requires that the PTLR methodology describes the reactor vessel material surveillance program.

As stated in the LAR, the reactor vessel material surveillance program for each unit is presented in the PTLR. Limerick Units 1 and 2 PTLRs are documented in Attachments 6 and 7 to the LAR dated September 20, 2020, respectively The NRC staff noted that the proposed P-T curves in the PTLRs are constructed using data from BWRVIP ISP, which is described in BWRVIP-78, “BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan” (ADAMS Accession No. ML003704011), and BWRVIP-86, Revision 1-A, “BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program Implementation Plan” (ADAMS Accession Nos. ML023190487 and ML042800398, respectively). By letter dated February 1, 2002 (ADAMS Accession No. ML020380691), the NRC approved the BWRVIP ISP.

Appendix A of the Limerick Units 1 and 2 PTLRs describes the following reactor vessel materials surveillance program at Units 1 and 2.

Unit 1 PTLR

The licensee installed three material surveillance capsules in the Unit 1 RPV in accordance with 10 CFR Part 50, Appendix H. The licensee stated that it has not withdrawn any of the Limerick Unit 1 surveillance capsules from the RPV. The licensee further stated that Limerick Unit 1 commitment to use the BWRVIP ISP in place of its existing surveillance programs is discussed in the NRC safety evaluation issued with License Amendment No. 167, Revision to the Reactor Pressure Vessel Material Surveillance Program (ADAMS Accession No. ML032310540), November 4, 2003. In addition, the licensee indicated that it will use the BWRVIP ISP during the period of extended operation. This commitment is discussed in the NRC safety evaluation for the Limerick license renewal application dated January 10, 2013 (ADAMS Accession No. ML12354A349) and Section 5.3.1.6, “Materials Surveillance,” of the Limerick UFSAR, R17 and is controlled by 10 CFR 50.59 requirements.

The licensee explained that under the BWRVIP ISP, Limerick Unit 1 is not a host plant, and no capsules are scheduled for removal from the Unit 1 RPV. The licensee further explained that representative surveillance capsule materials for the Limerick Unit 1 target beltline weld are contained in the River Bend Plant Capsules and Supplemental Surveillance Program (SSP) Capsules C, F, and H. Also, representative materials for the Limerick Unit 1 target beltline plate are in the Peach Bottom Atomic Power Station (Peach Bottom) Unit 2 Capsules. According to the licensee, at present, the next ISP surveillance capsule from River Bend is scheduled to be withdrawn and tested in approximately year 2025, and one additional capsule will be withdrawn during the license renewal period in approximately year 2030. In addition, one ISP capsule is scheduled to be withdrawn from Peach Bottom Unit 2 during the license renewal period in approximately year 2030.

As stated in Section 5.3.1.6, Materials Surveillance, of the Limerick UFSAR, R17, the licensee stated that in 2003, the NRC approved both Limerick units participation in the BWRVIP ISP as described in BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," and BWRVIP-86-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP)," Final Report, October 2002. By letter dated February 1, 2002, the NRC approved the use of BWRVIP-86. The NRC staff notes that BWRVIP-78 is superseded by BWRVIP-86-A. The NRC approved Limerick to participate in the ISP as documented in letter from S. P. Wall (U.S. NRC) to J. L. Skolds (Exelon Generation Company, LLC), "Limerick Generating Station Units 1 and 2 - Issuance of Amendment Re: Revision to the Reactor Pressure Vessel Material Surveillance Program (TAC Nos. MB7003 and MB7004)," dated November 4, 2003."

The NRC staff finds that Limerick Unit 1 PTLR satisfied Criterion 2 of GL 96 03 because the surveillance program in the Unit 1 PTLR follows the BWRVIP ISP in accordance with BWRVIP-86-A, as required by Section 5.3.1.6, Materials Surveillance, of the Limerick UFSAR, R17.

Unit 2 PTLR

The licensee installed three material surveillance capsules in the Unit 2 RPV. The licensee removed a surveillance capsule situated at the 120 degree (°) location of the RPV in October 2017 when the capsule holder was found damaged during a maintenance outage. The licensee did not reinstall the capsule. Therefore, two surveillance capsules remain on standby for use as backup capsules in accordance with the BWRVIP ISP in accordance with BWRVIP-86-A, or as otherwise needed.

The licensee stated that Limerick Unit 2 will use the BWRVIP ISP in place of its existing surveillance programs as discussed in the license amendment approved and issued by the NRC, Unit 2 License Amendment No. 130, Revision to the Reactor Pressure Vessel Material Surveillance Program (ADAMS Accession No. ML032310540), November 4, 2003. The licensee stated that Limerick Unit 2 will use the BWRVIP ISP in accordance with BWRVIP-86-A during the period of extended operation.

The licensee stated that under the ISP, Limerick Unit 2 is not a host plant and no further capsules are scheduled for removal from the Unit 2 RPV. The licensee stated that representative surveillance capsule materials for the Limerick Unit 2 target beltline weld are contained in the River Bend plant Capsules and Supplemental Surveillance Program Capsules C, F, and H. Also, representative materials for the Limerick Unit 2 target beltline plate

are in the Duane Arnold Energy Center (Duane Arnold) plant and Supplemental Surveillance Program-F surveillance capsules.

As discussed above, River Bend plans to withdraw and test its ISP capsules in 2025 and 2030. The licensee noted that one ISP capsule will be withdrawn from Duane Arnold upon plant closure in late 2020 or early 2021.

The NRC staff finds that the Limerick Unit 2 PTLR satisfied Criterion 2 of GL 96 03 because the surveillance program in the Unit 2 PTLR follows the BWRVIP ISP in accordance with BWRVIP-86-A as required by Section 5.3.1.6, Materials Surveillance, of the Limerick UFSAR, R17.

Criterion 3 requires that the PTLR methodology describes how the low-temperature overpressure protection (LTOP) system limits are calculated applying system/thermal hydraulics and fracture mechanics.

The NRC staff noted that LTOP system limits are applicable to pressurized water reactors only; Limerick Units 1 and 2 are of the BWR design. Therefore, Criterion 3 is not applicable to the Limerick PTLRs.

Criterion 4 requires that the PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2.

Section 4 of the Limerick Units 1 and 2 indicates that the ART values for the limiting beltline materials are calculated in accordance with RG 1.99, Revision 2. The NRC staff performed an independent calculation and verified that the licensee's ARTs for the limiting materials were calculated in accordance with RG 1.99, Revision 2. Therefore, the NRC staff finds that Limerick Units 1 and 2 PTLRs have satisfied Criterion 4.

Criterion 5 requires that the PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on the ASME Code, Section XI, Appendix G, and SRP Section 5.3.2.

Section 3 of the Limerick Units 1 and 2 PTLRs states that the P-T limits are calculated in accordance with NRC-approved BWROG-TP-11-022-A, Revision 1 which is based on the ASME Code, Section XI, Appendix G, and SRP Section 5.3.2. The NRC staff verified that the P-T limits in the Limerick Units 1 and 2 PTLRs were constructed in accordance with the ASME Code, Section XI, Appendix G, and SRP Section 5.3.2 as discussed further in this SE. Therefore, the NRC staff finds that Limerick Units 1 and 2 PTLRs satisfied Criterion 5.

Criterion 6 requires that the PTLR methodology describes how the minimum temperature requirements such as minimum bolt-up temperature and hydrotest temperature in Appendix G to 10 CFR Part 50 are applied to P-T curves.

Section 4 of the Limerick Units 1 and 2 PTLRs describes how the minimum temperature limits are set in accordance with Appendix G to 10 CFR Part 50. The NRC staff verified that the Unit 1 PTLR and Unit 2 PTLR describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to the P-T curves. Therefore, the NRC staff finds that Limerick Units 1 and 2 PTLRs satisfied Criterion 6.

Criterion 7 requires that the PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

Unit 1 PTLR

In accordance with Appendix A of BWROG-TP-11-022-A, Revision 1, the licensee reviewed the Unit 1 representative weld and plate surveillance materials data from the topical report, BWRVIP-135, Revision 3 (ADAMS Accession No. ML16076A209). The NRC staff determined that the Unit 1 PTLR is consistent with the procedure in BWRVIP-86, Revision 1-A with respect to the use of BWRVIP ISP capsule surveillance data in the ART calculation, and therefore finds that Unit 1 PTLR satisfied Criterion 7.

Unit 2 PTLR

In accordance with Appendix A of BWROG-TP-11-022-A, Revision 1, the licensee reviewed the Unit 2 representative BWRVIP ISP weld and plate surveillance materials data from the topical report, BWRVIP-135, Revision 3. The NRC staff determined that the Unit 2 PTLR is consistent with the procedure in BWRVIP-86, Revision 1-A, with respect to the use of BWRVIP ISP capsule surveillance data in the ART calculation, and therefore finds that Unit 2 PTLR satisfied Criterion 7.

As discussed above, the NRC staff finds that the Limerick Units 1 and 2 PTLRs have satisfied all seven criteria in of Attachment 1 to GL 96-03.

Therefore, the proposed changes require that the PTLR establish Limerick Unit 1 and 2, RCS P-T Limits for LCO 3.4.6.6 and SR 4.4.6, using the BWROG methodology and be submitted to the NRC upon issuance, which provides recordkeeping and reporting requirements necessary to assure operation of the facility in a safe manner, as required by 10 CFR 50.36(c)(5).

4.3 P-T Limits Evaluation

The NRC staff performed independent calculations to verify the proposed P-T limit curves using BWROG-TP-11-022-A, Revision 1; BWROG-TP-11-023-A, Revision 0; the ASME Code, Section XI, Appendix G; SRP 5.3.2; and 10 CFR 50, Appendices G and H. Specifically, the NRC staff verified the temperature rate limits, ART calculations, and P-T limit curves as discussed below.

4.3.1 Temperature Limits

Limerick Units 1 and 2 PTLRs specify the following temperature rates and limits as follows:

- Heat-up/cool-down rate limit during hydrostatic leak testing for Class 1 items: ≤25 ˚F/hour.
- Heat-up/cool-down rate limit during normal operating: ≤ 100 ˚F/hour.
- RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during recirculation pump startup: ≤ 145 ˚F.
- Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during recirculation pump startup: ≤ 50 ˚F.
- For Unit 1 PTLR and Unit 2 PTLR, RPV flange and adjacent shell temperature limit are ≥ 80 ˚F and ≥ 70 ˚F, respectively.

The NRC staff finds the above temperature rates and limits acceptable because they are consistent with the limits in typical BWR P-T curves and are consistent with those used in the licensee's P-T curve calculations.

4.3.2 Adjusted Nil-Ductility Transition Reference Temperature

The NRC staff notes that a typical P-T curve is constructed based on the fracture toughness of the RPV material resisting a postulated flaw with a depth of ¼ T (T = RPV shell wall thickness) at the inside surface (i.e., at the ¼ T location) and a postulated flaw at the outside surface (i.e., at the ¾ T locations) of the RPV shell wall thickness. As such, the adjusted nil-ductility transition reference temperature, ART, of the RPV material is calculated at the ¼ T and ¾ T locations as specified in RG 1.99, Revision 2. However, the licensee calculated the ART at the 1/4T location only because BWROG TP 11 022 A considers that the thermal gradient stresses at the ¼T location are assumed to be tensile for both heat-up and cool-down operation. The licensee explained that this approach is conservative because irradiation effects cause the allowable toughness at the ¼ T location to be less than that at the ¾ T location for a given metal temperature. The NRC staff performed an independent calculation based on the licensee's approach and confirmed that the licensee's approach is conservative because the licensee's methodology results in higher reactor coolant pressures and temperatures that the operators have to meet during the normal operation to minimize embrittlement in the reactor vessel materials. Therefore, the NRC staff finds the licensee's simplified approach results in conservative P-T curves and, therefore, is acceptable.

As part of P-T curves, the licensee also presented the initial RTNDT of the bottom head, feedwater nozzle and closure flange. These components are not affected by the neutron irradiation because they are not located in the high irradiation region. As such, the licensee used initial RTNDT for these components in lieu of ART in the P-T curve calculations.

For Limerick Unit 1, the NRC staff verified that the limiting beltline material is lower intermediate RPV shell plate No. 2, ID 17-2, heat number C7677-1 with an ART of 72.3 ˚F that is projected to 57 EFPY. Based on the fluence attenuation equation in RG 1.99, Revision 2, the NRC staff verified that the ART value for the LPCI and instrument nozzles are 59.9 °F and 47.6 ˚F, respectively. The NRC staff verified that the bottom head region, feedwater nozzle, and closure flange have a RTNDT of 12 ˚F, 48 ˚F, and 20 ˚F, respectively.

For Limerick Unit 2, the NRC staff verified that the limiting beltline material is the lower shell plate No. 1, ID 14-2, heat number B-3416-1, with an ART of 100.6 ˚F. In addition, the NRC staff verified that the ART for the LPCI nozzle and instrument nozzles are 60.2 ˚F and 37.7 ˚F, respectively. The NRC staff verified that the bottom head, feedwater nozzle, and closure flange have a RTNDT of 28 ˚F, 46 ˚F, and 10 ˚F, respectively.

The NRC staff noted that for Limerick Unit 1, the ART for the limiting beltline plate C7677-1 is 72.3 ˚F in the proposed PTLR, but it is reported as 74 ˚F in Table 4.3.2-1 of the Limerick license renewal application dated June 22, 2011 (ADAMS Accession No. ML11179A101). For Limerick Unit 2, the ART for the limiting beltline plate B3416-1 is 100.6 ˚F in the proposed PTLR whereas it is reported as 102 ˚F in Table 4.2.3-3 of the Limerick license renewal application. The NRC staff finds that the difference in the ARTs between the proposed PTLR and the license renewal application is caused by the difference in the neutron fluence used. The licensee stated that based on information from the previous fuel cycles and new fuel used, it revised the neutron fluence projection for the 57 EFPY as shown in Attachment 14 to the September 29, 2020,

letter. The NRC staff finds that the difference in the ARTs is acceptable because it is not significant and is caused by the revised neutron fluence projections for 57 EFPY.

The NRC staff finds that the ART values for the limiting materials are acceptable because the licensee calculated the ARTs based on RG 1.99, Revision 2 and the BWRVIP ISP data.

4.3.3 P-T Curves

The NRC staff evaluated each of Curves A, B, and C which contains three sub-category curves. The sub-category curves are the beltline region curve, non-beltline region curve, and bottom head region curve. In addition, Curves A, B, and C contain the bounding curve representing the beltline region, non-beltline region and bottom head region as discussed below.

4.3.3.1 RPV Beltline Region Curve

The licensee stated that its P-T curves are developed to bound all ferritic materials in the RPV, including the structural discontinuities such as nozzles. The licensee further stated that the RPV beltline region includes the instrument nozzles, the LPCI nozzles, and the beltline shell plates and welds. For the beltline region, the licensee constructed three P-T curves-- one for the limiting beltline component (e.g., shell plate), one for the LPCI nozzle, and one for the instrument nozzle. From these three P-T curves, the licensee selected a single bounding P-T curve representing the beltline region to be used in Curves A, B, and C.

To construct the P-T curve for the limiting beltline shell plate, the licensee derived the maximum K_{ip}, and K_{it}, used the ART, and constructed the curve in accordance with equations in BWROG-TP-11-022-A, Revision 1.

To construct the P-T curve for the LPCI nozzle, the licensee used a 3-dimensional symmetric one-quarter size finite element model that represents both units, as the LPCI nozzle geometry is identical for both units. The licensee postulated a 1/4T wall thickness flaw located at the LPCI nozzle blend radius, derived the maximum K_{ip}, and K_{it}, and constructed the P-T curve for the LPCI nozzle in accordance with BWROG-TP-11-022-A, Revision 1. The licensee used the temperature-dependent material properties from the ASME Code, Section III, 1968 Edition with 1969 Summer Addenda and Article 4 of the 1969 Winter Addenda and from the ASME Code, Section II, Part D, 2001 Edition with Addenda through 2003. The Limerick Units 1 and 2 reactor vessels are made of SA-533, Grade B material.

The licensee stated that the instrument nozzles are fabricated from nonferritic materials and do not require evaluation for loss of fracture toughness. As such, the licensee used the empirical equations in BWROG-TP-11-023-A, Revision 0 to derive K_{ip}, and K_{it}, and constructed the P-T curve for the instrument nozzle.

The NRC staff evaluated the P-T curves for the beltline shell plate, LPCI nozzle and instrument nozzle. The NRC staff verified that the P-T curve representing the beltline region in Curves A, B and C for the Unit 1 PTLR and Unit 2 PTLR are acceptable because (1) it was constructed based on BWROG-TP-11-022-A, Revision 1, and BWROG-TP-11-023-A, Revision 0, and (2) it bounds curve from the P-T curves for the beltline shell plate, LPCI nozzle, and Instrument nozzle.

4.3.3.2 RPV Non-Beltline Region Curve

Appendix G to 10 CFR Part 50 requires that the P-T limit curves be developed for the ferritic materials not in the RPV beltline region. In addition, RIS 2014-11 clarifies that P-T curve calculations for ferritic RPV components that are not beltline shell materials may define P-T curves that are more limiting than those calculated for the RPV beltline shell materials.

The licensee stated that the P-T curve for the non-beltline region is represented by the feedwater nozzles because they are the limiting component. The licensee used finite element analyses to develop the thermal and pressure stress distributions in a feedwater nozzle, from which K_{ip}, and K_{it} were derived to construct the P-T curve in accordance with BWROG-TP-11-022-A, Revision 1. The licensee considered the temperature and pressure in the power uprated conditions as part of stress analysis.

The NRC staff notes that RPV closure flange is also part of non-beltline region that must be considered in the P-T curves in accordance with 10 CFR Part 50, Appendix G. The NRC staff finds that the bounding P-T curve representing the non-beltline region in Curves A, B, and C for Limerick Units 1 and 2 PTLRs are acceptable because it was constructed based on BWROG-TP-11-022-A, Revision 1, and it includes the P-T limits from the limiting materials (i.e., feedwater nozzles and closure flange).

4.3.3.3 RPV Bottom Head Region Curve

For the bottom head region, the licensee calculated K_{ip}, and K_{it}, and constructed the P-T curves based on BWROG-TP-11-022-A, Revision 1. The NRC staff finds that the bounding P-T curve representing the bottom head region in Curves A, B, and C for the Limerick Units 1 and 2 PTLRs is acceptable because it is appropriately constructed based on BWROG-TP-11-022-A, Revision 1.

4.3.3.4 <u>Curve A</u>

The licensee stated that Curve A is for the hydrostatic pressure and leak test; therefore, the thermal stress is considered negligible. As such, the licensee set the stress intensity factor, K_{it} to zero. In addition, the licensee limited the heat-up or cool-down rate during hydrostatic leak test to be ≤ 25 ˚F/hour. The NRC staff finds it acceptable to set K_{it} to zero because temperature gradient in the RPV shell wall is too small to produce any significant K_{it} during the hydrostatic and leakage tests. As specified in BWROG-TP-11-022-A, Revision 1, for Curve A, the licensee set the safety factor for the pressure stress intensity factor, K_{ip} to 1.5.

Item 1.a in 10 CFR 50, Appendix G, Table 1, requires that when the operating pressure is ≤ 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload. The NRC staff verified that the proposed Curve A in the Unit 1 PTLR and Unit 2 PTLR satisfied Item 1.a.

Appendix G to 10 CFR Part 50, Table 1, “Pressure and Temperature requirements for the Ractor Pressure Vessel,” Item 1.b requires that when the pressure is > 20 percent of the preservice system hydrostatic test pressure (1,563 pounds per square inch gauge, psig), the minimum temperature must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload plus 90 ˚F. In addition, the Item 1.b requirement creates a notch (or bend) in the non-beltline P-T curve at 312.6 psig (20 percent of

1,563 psig) because the closure flange is part of the non-beltline region. The NRC staff verified that the proposed Curve A in the Limerick Units 1 and 2 PTLRs satisfied Item 1.b.

The NRC staff finds that the overall Curve A in Figure 1 of the Limerick Units 1 and 2 PTLRs is acceptable because it satisfies Items 1.a and 1.b in 10 CFR Part 50, Appendix G, and bounds the maximum P-T curves of the bottom head region, non-beltline region and beltline region.

4.3.3.5 Curve B

The NRC staff verified that as specified in BWROG-TP-11-022-A, Revision 1, to construct Curve B, the licensee used the safety factor of 2.0 and 1.0 for the pressure and thermal stress intensity factor, respectively.

With regard to the required minimum temperature, 10 CFR Part 50, Appendix G, Table 1, Item 2.a requires that when the operating pressure is ≤ 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload. The NRC staff verified that the proposed Curve B in the Limerick Units 1 and 2 PTLRs has satisfied Item 2.a.

Appendix G to 1- CFR Part 50, Table 1, Item 2.b, requires that when the pressure is greater than (>) 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload plus 120 ˚F. The Item 2.b requirement creates a notch (or bend) for the non-beltline P-T curve at 312.6 psig because the closure flange is part of the non-beltline region. The NRC staff verified that the proposed Curve B in the Limerick Units 1 and 2 PTLRs has satisfied Item 2.b.

The NRC staff noted that Curve B for Limerick Units 1 and 2 contains a minimum temperature for the beltline region of 82.5 ˚F and 82.8 ˚F, respectively. The NRC staff verified that these two minimum temperatures are derived based on the RPV pressure set to zero. The NRC staff noted that this minimum temperature is not part of requirements in 10 CFR Part 50, Appendix G; however, these temperatures are appropriately included in the P-T curves to ensure that the RPV material is protected from embrittlement even when the RPV pressure is zero.

The NRC staff finds that the overall Curve B in Figure 2 of the Limerick Units 1 and 2 PTLRs is acceptable because it satisfies Items 2.a and 2.b in 10 CFR Part 50, Appendix G and bounds the maximum P-T curves of the bottom head region, non-beltline region and beltline region.

4.3.3.6 Curve C

Appendix G to 10 CFR Part 50, Table 1, Items 2.c, 2.d, and 2.e require that the P-T curves for the core critical condition (Curve C) be 40 ˚F greater than the P-T limit curves for the core not critical (Curve B) under all operating pressure conditions. The NRC staff verified that the bottom head region curve, non-beltline region curve and beltline region curve in Curve C are 40 ˚F more than the corresponding curves in Curve B. The NRC staff finds that the proposed Curve C in the Limerick Units 1 and 2 PTLRs satisfied the 40 ˚F requirement of Items 2.c, 2.d, and 2.e.

Appendix G to 10 CFR Part 50, Table 1, Item 2.c requires that when the operating pressure is ≤ 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be larger of the minimum permissible temperature for the inservice system hydrostatic pressure

test or the closure flange RTNDT plus 40 ˚F. The NRC staff verified that the proposed Curve C in the Limerick Units 1 and 2 PTLRs has satisfied Item 2.c.

Appendix G to 10 CFR Part 50, Table 1, Item 2.d requires that when the pressure is > 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be larger of the minimum permissible temperature for the inservice system hydrostatic pressure test or closure flange RTNDT plus 160 ˚F. The NRC staff finds that there is a notch (or bend) in the non-beltline P-T curve at 312.6 psig. The NRC staff verified that the proposed Curve C in the Limerick Units 1 and 2 PTLRs has satisfied Item 2.d.

Appendix G to 10 CFR Part 50, Table 1, Item 2.e requires that for BWRs, 60 ˚F be added to the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload when the operating pressure is less than 20 percent of preservice system hydrostatic test pressure. The NRC staff verified that the proposed Curve C in the Limerick Units 1 PTLR and 2 PTLRs has satisfied Item 2.e.

The NRC staff finds that the overall Curve C in Figure 3 of the Limerick Units 1 PTLR and 2 PTLRs is acceptable because it satisfies Items 2.c, 2.d, and 2.e in 10 CFR Part 50, Appendix G and bounds the maximum P-T curves of the bottom head region, non-beltline region, and beltline region.

4.3.3.7 Composite P-T Limit Curves

The NRC staff verified that the composite P-T curves in Figure 4 of the Limerick Units 1 PTLR and 2 PTLRs consist of curves representing the hydrostatic pressure and leak test condition (Curve A), the normal operation with core not critical condition (Curve B), and the normal operation with core critical condition (Curve C). The NRC staff further verified that Curves A, B, and C in Figure 4 of the Limerick Units 1 PTLR and 2 PTLRs contain the bounding curves from the bottom head region, non-beltline region and beltline region. Therefore, the NRC staff finds that the final composited P-T Curves A, B, and C in Figure 4 of the Limerick Units 1 and 2 PTLRs are acceptable because they satisfy the fracture toughness provisions in 10 CFR Part 50, Appendix G, and the surveillance program provisions in 10 CFR Part 50, Appendix H.

4.3.4 P-T Limits

To evaluate the proposed P-T limits for Limerick Units 1 and 2, the staff reviewed the PTLR submittal as well as the fluence calculations. The staff reviewed the selection of materials, the ART values used, as well as the adherence to approved methodologies.

The licensees calculation stated that the P-T limits were derived by: using BWROG LTR BWROG-TP-11-022-A (SIR-05-044 Revision 1-A) neutron fluence calculated in accordance with RG 1.190 using the RAMA computer code as approved by the NRC. The ART values for the limiting beltline materials were calculated in accordance with RG 1.99.

The proposed P-T limits in the PTLR were generated using the BWROG report methodology based on ASME B&PV Code, Section XI, Appendix G. The curves are developed for three categories of operation: hydrostatic pressure tests and leak tests, core not critical operation, and core critical operation. The licensee used minimum temperature limits in accordance with 10 CFR Part 50, Appendix G.

The adjusted reference temperature of the limiting beltline material is used to adjust beltline P-T curves to account for irradiation effects. Limerick discusses the use of RG 1.99 to determine a chemistry factor for welds and for plates and forgings. The Cu and Ni values that were used were obtained from the evaluation of vessel plate, weld, and forging materials. The licensee used fluence values calculated in the supplied fluence evaluation to come up with a limiting value for ART. The ART value for beltline plates and welds that was found to be limiting is 100.6 ˚F. Limerick has two sets of nozzles in the RPV beltline. The instrument nozzles and the low pressure coolant injection nozzles were evaluated to ensure that the P-T limits are bounding. The Limiting ART value for the LPCI nozzles and welds was found to be 60.2 ˚F. The limiting ART value for the instrument nozzles and welds was found to be 37.7 ˚F.

During the P-T curve development the ¼ T and ¾ T locations are looked at for a given EFPY. Limerick simplified the approach to consider tensile stress for both heat up and cool down at the ¼ T location. This approach is conservative because at the ¼ T location the irradiation effects cause the allowable toughness to be less than that at the ¾ T location.

The staff verified that the proposed P-T limits are consistent with the requirements in 10 CFR Part 50, Appendix G, for the minimum temperature of the closure flange regions. For all proposed P-T limit curves, the far left straight line corresponds to the minimum bolt up temperature. For Limerick Unit 1, this is 80 ˚F and for Limerick Unit 2, it is 70 ˚F. Table 1 of Appendix G to 10 CFR Part 50 requires different minimum temperatures for P-T limits depending on whether the pressure is less than or greater than 20 percent of the preservice hydrostatic test pressure. This requirement creates a notch in the P-T curves. In the proposed P-T curves a notch is observed for the non-beltline P-T limits because this region contains the closure flange. For the non-beltline P-T limits the NRC staff verified that the requirements of Table 1 of Appendix G to 10 CFR Part 50 are met by confirming the notch temperatures of 110 ˚F and 100 ˚F for Limerick Units 1 and 2 respectively for Curve A. For Curve C the temperatures of 180 ˚F and 170 ˚F were verified. Therefore, the staff determined that the proposed P-T limits meet the minimum temperature requirements listed in Table 1 of Appendix G to 10 CFR Part 50.

Ferritic RCPB Components Outside of the RPV

The SE for the BWROG report requires licenses to confirm that all ferritic RCPB components that are not part of the RPV will not define a more restrictive operating temperature than the proposed P-T limits. The licensee stated in the LAR that it confirmed that the lowest service temperatures for all ferritic RCPB components that are not part of the RPV are below the lowest operating temperature in the proposed P-T limits. The confirmation is included in the Operating Limits section of the Limerick Units 1 and 2 PTLRs. The NRC staff concludes that the lowest service temperatures for all ferritic RCPB components that are not part of the RPV are below the lowest operating temperature in the proposed P-T limits. Therefore, the NRC staff finds that the revised P-T limit curves satisfy this criterion and are acceptable.

4.4 Conformance with TSTF-419-A

The NRC staff reviewed the proposed changes to Limerick Units 1 and 2 TSs to determine whether they conform to TSTF-491-A, which modified TSTF-419, Revision 0.

The NRC staff initially approved TSTF-419, Revision 0, by letter dated March 21, 2002. By letter dated August 4, 2011, the NRC staff modified that approval by clarifying that requests for amendments to implement TSTF-419, Revision 0, need to include a full citation to the

NRC-approved topical reports used in the PTLR methodology, including the revision number and date of the topical report.

The staff noted that Limerick Units 1 and 2 TSs are formatted according to a precursor to the current format of NUREG-1433 (ADAMS Accession Nos. ML12104A192 (Volume 1) and ML21204A193 (Volume 2), but the differences in format and numbering between Limerick TSs and NUREG-1433 STSs do not affect the applicability of TSTF-419 to Limerick TSs.

Based on its review of the proposed changes to Limerick TSs, the NRC staff finds that the proposed revisions to TS Section 1.0 define the PTLR and proposed revisions to TS Section 6.0 require that the PTLR establish RCS P-T Limits for Unit 1 and 2 LCO 3.4.6 and SR 4.4.6, using the BWROG methodology, which is properly referenced as BWROG-TP-11-022-A, Revision 1, dated August 2013. The new TS 6.9.1.13 requires that the PTLR be submitted to the NRC upon issuance and LCO 3.4.6 and SR 4.4.6 are revised to indicate that limits are specified in the PTLR. The NRC staff also finds the proposed TS changes appropriately adopt TSTF-419 and in a format compatible with format of the Limerick Unit 1 and 2 TSs. The NRC staff also finds the proposed changes require that the PTLR establish Unit 1 and 2, RCS P-T Limits for LCO 3.4.6. and SR 4.4.6, using the BWROG methodology and be submitted to the NRC upon issuance, which provides recordkeeping and reporting requirements necessary to assure operation of the facility in a safe manner, as required by 10 CFR 50.36(c)(5).

5.0 TECHNICAL CONCLUSION

Based on information submitted, the NRC staff determined that (1) the proposed P-T curves in the Limerick Units 1 and 2 PTLRs considered all ferritic RPV materials, (2) the proposed P-T curves are constructed based on the methodology in BWROG-TP-11-022-A, Revision 1; BWROG-TP-11-023-A, Revision 0; SRP 5.3.2; and the ASME Code, Section XI, Appendix G, and (3) the proposed Limerick Units 1 and 2 PTLRs satisfy applicable requirements in 10 CFR Part 50, Appendices G and H, 10 CFR 50.60, GL 96-03, and TSTF-419-A.

The NRC staff finds that the proposed Limerick Units 1 and 2 PTLRs are consistent with GL 96-03 with respect to PTLR implementation and therefore, is approved as part of the Limerick Units 1 and 2 licensing basis. The NRC staff further concludes that the proposed P-T limits valid for 57 EFPY are based on an acceptable methodology documented in BWROG-TP-11-022-A. The staff reviewed the licensee calculations and determined that the P-T limits are developed appropriately using BWROG report methodology and satisfy the requirements of Appendix G to Section XI of the ASME B&PV Code and Appendix G to 10 CFR Part 50. Therefore, the NRC staff concludes that the proposed Unit 1 PTLR and Unit 2 PTLR are acceptable for use up to 57 EFPY to provide the safe operation of the RPV at Limerick Units 1 and 2.

The NRC staff determined the addition of the PTLR to the Administrative Controls will contain the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner, as required by 10 CFR 50.36(c)(5). In addition, the revised LCOs will continue to specify the lowest functional capability or performance levels of equipment required for safe operation of the facility, as required by 10 CFR 50.36(c)(2). The SRs will continue to assure that the necessary quality of systems and components is maintained that facility operation will be within safety limits, and that the limiting conditions for operation will be met, as required by 10 CFR 50.36(c)(3). Therefore, the NRC staff concludes that the TS, as revised by the

proposed changes, will continue to meet the requirements of 10 CFR 50.36(c)(2), (c)(3), and (c)(5).

The NRC staff finds that plant operation continues to be limited in accordance with the requirements of Appendix G to 10 CFR Part 50 and that the P-T limits in the TSs are established using a methodology approved by the NRC. Therefore, the NRC staff concludes that the proposed changes continue to meet the requirements of 10 CFR 50.36.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on June 25, 2021. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 published in the *Federal Register* on December 1, 2020 (85 FR 77274). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 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.

Principal Contributors: J. Tsao, NRR
C. Fairbanks, NRR
J. Miller, NRR
M. Hamm, NRR

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SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 − ISSUANCE OF AMENDMENT NOS. 253 AND 215 RE: TECHNICAL SPECIFICATION CHANGES RELATED TO RELOCATION OF PRESSURE AND TEMPERATURE LIMIT CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2020-LLA-0221) DATED SEPTEMBER 28, 2021

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OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LAiT	NRR/DORL/LPL2-1/LA
NAME	VSreenivas	KZeleznock	KGoldstein
DATE	06/25/2021	07/01/2021	07/19/2021
OFFICE	NRR/DNRL/NVIB/BC(A)	NRR/DSS/SNRB/BC	NRR/DSS/STSB/BC (A)
NAME	DWidrewitz	RKarus	NJordan
DATE	04/01/2021	03/30/2021	07/10/2021
OFFICE	OGC-NLO	NRR/DORL/LPL1/BC	NRR/DORL/LPL1/PM
NAME	MYoung	JDanna	VSreenivas
DATE	09/03/2021	09/28/2021	09/28/2021

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