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

December 14, 2017

Mr. Peter P. Sena, III
President and Chief Nuclear Officer
PSEG Nuclear LLC – N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT TO REVISE AND RELOCATE THE PRESSURE-TEMPERATURE LIMIT CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT (CAC NO. MF9502; EPID L-2017-LLA-0204)

Dear Mr. Sena:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 209 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 27, 2017, as supplemented by letters dated April 28, 2017, and September 5, 2017. The amendment changes the TSs to relocate the reactor coolant system pressure-temperature (P-T) limit curves from the TSs to a new licensee-controlled document called the Pressure and Temperature Limits Report. The amendment also revises the 32 effective full power years P-T limit curves and approves P-T limit curves applicable through the license renewal term. The revisions to the curves were required due to the results of a recently pulled and tested reactor pressure vessel surveillance capsule.

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

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 209 to Renewed License No. NPF-57
2. Safety Evaluation

cc w/Enclosures: Distribution via Listserv

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

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
Renewed License No. NPF-57

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

   A. The application for amendment filed by PSEG Nuclear LLC dated March 27, 2017, as supplemented by letters dated April 28, 2017, and September 5, 2017, 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 set forth in 10 CFR Chapter I;

   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 Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

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

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: December 14, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 209

HOPE CREEK GENERATING STATION

RENEWED FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

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
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 marginal lines indicating the areas of change.

Remove	Insert
ii	ii
xi	xi
xviii	xviii
xxiv	xxiv
1-5	1-5
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
3/4 4-23	3/4 4-23
3/4 4-23a	3/4 4-23a
3/4 4-23b	3/4 4-23b
6-20	6-20
6-21	6-21

reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) PSEG Nuclear LLC, 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;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 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

(6) PSEG Nuclear LLC, 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. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

(7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

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 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

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 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

1.30 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 AND TEMPERATURE LIMITS REPORT (PTLR)

1.30-1 The PTLR is the 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 6.9.1.10.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.32 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. Each 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.

REACTOR COOLANT SYSTEM

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 limits specified in the PTLR,

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

c. A maximum temperature change within 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 metal temperature shall be maintained within 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 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the curves specified in the PTLR.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to the limit specified in 3.4.6.1.d.

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

    1. ≤ 110°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.

Figure 3.4.6.1-1
DELETED

Figure 3.4.6.1-2
DELETED

Figure 3.4.6.1-3
DELETED

6.9.1.8 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the PSEG Nuclear LLC generated CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following Technical Specifications:

2.2	Reactor Protection System Instrumentation Setpoints
3/4.1.4.3	Rod Block Monitor
3/4.2.1	Average Planar Linear Heat Generation Rate
3/4.2.3	Minimum Critical Power Ratio
3/4.2.4	Linear Heat Generation Rate
3/4.3.1	Reactor Protection System Instrumentation
3/4.3.6	Control Rod Block Instrumentation

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC as applicable in the following document:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number title, revision, date, and any supplements).

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

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, 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)

6.9.1.10

a. 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:

   1. Limiting Condition for Operation Section 3.4.6, "RCS Pressure/Temperature Limits"

   2. Surveillance Requirement Section 4.4.6, "RCS Pressure/Temperature Limits"

b. 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 document:

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

c. 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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 When a report is required by Action 10 of Specification 3/4.3.1, “RPS Instrumentation,” a report shall be submitted within 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10 of the Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated March 27, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17086A364), as supplemented by letters dated April 28, 2017, and September 5, 2017 (ADAMS Accession Nos. ML17118A092 and ML17248A127, respectively), PSEG Nuclear, LLC (PSEG, the licensee) submitted a license amendment request (LAR) for Hope Creek Generating Station (Hope Creek). The amendment requested changes to the Hope Creek Technical Specifications (TSs) to relocate the reactor coolant system (RCS) pressure-temperature (P-T) limits from the TSs to a new licensee-controlled document called the Pressure and Temperature Limits Report (PTLR). The LAR also proposed revisions to the 32 effective full power years (EFPY) P-T limit curves and proposed new P-T curves applicable through 44 EFPY and 56 EFPY (i.e., through the end of the license renewal term). The revisions to the curves were required due to the results of a recently pulled and tested reactor pressure vessel (RPV) surveillance capsule. The proposed Hope Creek P-T limit curves to be incorporated in the PTLR were developed based on the U.S. Nuclear Regulatory Commission (NRC or the Commission)-approved methodologies described in the Boiling Water Reactor Owners' Group (BWROG) Topical Report BWROG-TP-11-022-A (SIR-05-044), Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," August 2013 (ADAMS Accession No. ML13277A557). This guidance will be referred to as BWROG-TP-11-022-A, Revision 1, or the BWROG PTLR methodology in this safety evaluation (SE).

The licensee's supplemental letter dated September 5, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

2.1 Technical Specification Requirements

The proposed amendment involves changes to the TS requirements for the Hope Creek P-T limits. Section 3/4.4.6, "Pressure/Temperature Limits," of the Hope Creek TSs contains the TS requirements for operation of the Hope Creek RCS in accordance with the P-T limits. The P-T limits are traditionally established in the TSs to protect the integrity of the reactor coolant pressure boundary (RCPB) from rapidly propagating fracture during normal operating and pressure test conditions per the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," and 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." The ferritic RPV materials have lower toughness at low temperatures compared to normal operating temperature. Therefore, acceptable operation of the RCS is defined by maintaining RCS pressure less that the P-T limits and RCS temperature greater than the P-T limits for all modes of reactor operation when the RPV closure head is tensioned to the vessel.

The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." The requirements for TS content in 10 CFR 50.36 include the following categories of plant safety criteria: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

2.2 Requirements for P-T Limits

Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC), to 10 CFR Part 50, GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 30, "Quality of Reactor Coolant Pressure Boundary," states that components that are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical.

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," states that the RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steady state, and transient stresses; and (4) size of flaws.

Section 50.60 of 10 CFR requires that all light-water nuclear power reactors meet the fracture toughness and material surveillance program requirements set forth in 10 CFR Part 50, Appendix G, and 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," in order to protect the integrity of the RCPB.

Appendix G to 10 CFR Part 50 establishes fracture toughness requirements to maintain the integrity of the RCPB in nuclear power plants. P-T limit requirements for the RPV are established in paragraph IV.A.2 and Table 1 of this rule. Paragraph IV.A.2 and Table 1 specify that P-T limit curves and minimum temperature requirements for the RPV are defined by the operating condition (i.e., pressure testing or normal operation, including anticipated operational occurrences), the RPV pressure, whether or not fuel is in the RPV, and whether the core is critical. In Table 1, the RPV pressure is defined as a percentage of the preservice system hydrostatic test pressure. The requirements for both the RPV P-T limit curves and the minimum RPV temperature must be met for all normal operating and pressure test conditions.

Paragraph IV.A.2 of 10 CFR Part 50, Appendix G, states that the P-T limits identified as "ASME Appendix G limits" in Table 1 of this rule require that the limits must be at least as conservative as those obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Code. The minimum temperature requirements given in Table 1 pertain to the controlling material at low temperatures for inservice plants (this is the material in the RPV closure flange region that is highly stressed by bolt preload). For inservice plants, the metal temperature of the controlling material in the closure flange region, which has the least favorable combination of stress and temperature, must exceed the applicable minimum temperature requirement for the operating condition and RPV pressure specified in Table 1.

Additionally, 10 CFR Part 50, Appendix G, requires that applicable surveillance data from RPV material surveillance programs be incorporated into the calculations of the P-T limits and that the P-T limits be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

NRC Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement on the low-alloy steels used for light-water RPVs. Additional guidance related to the NRC staff's review of P-T limit curve submittals is found in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock."

The ASME Code, Section XI, Appendix G, methodology for generating P-T limit curves is based upon the principles of linear elastic fracture mechanics. The fundamental parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state in the component and flaw configuration. The ASME Code, Section XI, Appendix G, requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal operating conditions, and a safety factor of 1.5 on these stress intensities for hydrostatic and pressure testing limits. The ASME Code, Section XI, Appendix G, specifies that the P-T limits shall be generated by postulating a flaw with a depth that is equal to one-quarter of the low alloy steel RPV section thickness (1/4T, where T is vessel beltline thickness) and a length equal to 1.5 times the RPV section thickness. The critical locations in the RPV section thickness for calculating the P-T limit curves are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface flaws and outside surface flaws, respectively.

The ASME Code, Section XI, Appendix G, specifies that P-T limit curve calculations shall be based, in part, on the reference nil-ductility transition temperature, RT_{NDT}, for the material. The RT_{NDT} is the fundamental parameter for defining the critical stress intensity factor (K_{IC}, also referred to as plane strain fracture toughness) for the material as a function of temperature. Appendix G to 10 CFR Part 50 requires that RT_{NDT} values for materials in the RPV beltline region be adjusted to account for the effects of neutron irradiation. RG 1.99, Revision 2, defines acceptable methodologies for calculating the adjusted RT_{NDT} (ART) due to neutron irradiation. The ART is defined as the sum of the initial (unirradiated) RT_{NDT}, the mean value of the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper (Cu) and nickel (Ni) in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the postulated flaw depths described above. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the Cu and Ni content, the neutron fluence, and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

Recent NRC staff guidelines published in NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," October 14, 2014 (ADAMS Accession No. ML14149A165), provide additional NRC staff expectations for evaluations of P-T limits in licensing applications and PTLRs, including specific guidance on the consideration of neutron fluence and structural discontinuities in the development of P-T limits.

To satisfy the requirements of 10 CFR 50, Appendix G, methods for determining fast neutron fluence are necessary to continuously monitor the fracture toughness of the RPV beltline materials during operation. Appendix H to 10 CFR Part 50 requires the installation of surveillance capsules, including material test specimens and neutron flux dosimeters, to provide data for material damage correlations as a function of neutron fluence. NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods acceptable to the NRC staff for determining the RPV neutron fluence with respect to meeting the regulatory requirements discussed above.

2.3 Acceptable Fluence Calculations

RG 1.190 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the GDC contained in Appendix A to 10 CFR Part 50. In consideration of the guidance set forth in RG 1.190, GDCs 14, 30, and 31, as described above, are applicable.

The guidance provided in RG 1.190 indicates that the following elements comprise an acceptable fluence calculation:

1. determination of the geometrical and material input data,
2. determination of the core neutron source,
3. propagation of the neutron fluence from core to vessel and into the cavity, and
4. qualification of the calculational procedure.

The NRC's review was performed to establish that elements 1 through 4 above of the calculational method adhere to the regulatory positions set forth in RG 1.190.

2.4 Criteria for PTLRs

On January 31, 1996, the NRC staff issued Generic Letter (GL) 96-03, "Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," to inform licensees that they may request a license amendment to relocate the P-T limits from the TS LCOs to a PTLR or other licensee-controlled document that would be governed by the TS administrative controls. In order to permit relocation of the P-T limits to a PTLR, GL 96-03 states that licensees shall generate their P-T limits in accordance with a methodology that was previously approved by the NRC staff based on their compliance with the requirements of 10 CFR Part 50, Appendices G and H. GL 96-03 also states that the NRC-approved PTLR methodology must be incorporated by reference in the administrative controls section of the TSs and that the PTLR be defined in Section 1.0 of the TSs. Attachment 1 to GL 96-03 provides a list of seven technical criteria that generic PTLR methodologies and plant-specific PTLR license amendment applications should satisfy in order to ensure compliance with the requirements of 10 CFR Part 50, Appendices G and H.

NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS PTLR," August 4, 2003, amended the Standard Technical Specifications (STS) for all domestic light-water reactor designs to: (1) delete references to the TS LCOs for the P-T limits in the TS definition of the PTLR, and (2) revise the standard administrative controls for the PTLR in STS Section 5.6 to allow NRC-approved topical reports for PTLR methodologies to be identified by number and title. TSTF-419-A did not change the requirement that the PTLR methodology be approved by the NRC or the TS requirement to operate the RCS within the limits specified in the PTLR. Any changes to the PTLR methodology referenced in the TS administrative controls would continue to require NRC staff review and approval pursuant to the license amendment application provisions of 10 CFR 50.90.

If a plant is still operating with custom TSs, the guidance of TSTF-419-A is acceptable for referencing in plant-specific licensing applications to implement a PTLR because the TSTF adequately addresses the specific TS changes needed to implement a PTLR, irrespective of whether the plant operates with custom TSs or standard TSs, and is consistent with the intent of GL 96-03.

The licensee proposes to implement the methodology contained in BWROG-TP-11-022-A, Revision 1. Regarding fluence, Table 1-1 of the PTLR methodology states, "[Neutron fluence is] not covered by this LTR [BWROG-TP-11-022-A, Revision 1]. Fluence methods and results must comply with RG 1.190 and have NRC approval for use with this LTR." Therefore, the topic of neutron fluence is a plant-specific action item that must be addressed by licensees proposing to implement BWROG-TP-11-022-A, Revision 1.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Licensee's Evaluation</u>

PSEG's LAR to implement a PTLR proposes the following changes to the Hope Creek TSs:

1. Adds a definition in TS Section 1.0, "Definitions," for the PTLR.

2. Revises the TS Index to reflect additions, deletions, and pagination changes.

3. Revises TS Section 3/4.4.6, "Pressure/Temperature Limits," LCOs and SRs as follows:

    - Revises LCO 3.4.6.1 for the RCS P-T limits to refer to the limits in the PTLR.

    - Revises LCO 3.4.6.1.a for the maximum heatup rate to refer to the limits specified in the PTLR.

- Revises LCO TS 3.4.6.1.b for the maximum cooldown rate to refer to the limits specified in the PTLR.
- Revises LCO 3.4.6.1.c for the maximum temperature change for hydrostatic and leak testing operations to refer to the limits specified in the PTLR.
- Revises LCO 3.4.6.1.d for the RPV flange and head flange metal temperature to refer to the limits specified in the PTLR when the RPV closure head bolting studs are under tension.

4. Revises SRs 4.4.6.1.1 and 4.4.6.1.2 to refer to the limits specified in the PTLR.

5. Revises SR 4.4.6.1.3 to refer to the P-T limit curves specified in the PTLR.

6. Removes the present P-T curves located in Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 of the TSs. Revised P-T curves (evaluated below) are placed in the proposed PTLR.

7. Adds a new Specification, TS 6.9.1.10, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to Subsection 6.9, "Reporting Requirements," in Section 6.0, "Administrative Controls." The new specification includes:

   - The individual TS LCOs and SRs that address reactor coolant system P-T limits.
   - The reference to the NRC-approved licensing topical report that documents the generic PTLR methodology.
   - The requirement that any revisions to the PTLR be submitted to the NRC.

The licensee stated that the relocation of the P-T limit curves to the PTLR adopts the methodology provided in BWROG-TP-11-022-A, Revision 1. The licensee also stated that the proposed TS changes are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419-A, which allows the licensee to relocate its P-T limit curves from the plant TSs to PTLRs. The licensee noted that in order to implement a PTLR, the analytical methods used to develop the P-T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the administrative controls section of the plant TSs.

The proposed PTLR for Hope Creek contains new P-T limit curves that are valid for peak RPV inner diameter (ID) fluence values of 8.81×10^{17} n/cm^2, 1.26×10^{18} n/cm^2 and 1.63×10^{18} n/cm^2 corresponding to 32, 44, and 56 EFPY of core operation, respectively. The licensee stated that the P-T limit curves were developed in accordance with the methodology in BWROG-TP-11-022-A, Revision 1. The licensee indicated that the purpose of the BWROG PTLR methodology is to provide boiling water reactors (BWRs) with an NRC-approved topical report that can be referenced in plant TSs to establish BWR fracture mechanics methods for generating P-T limits curves and other associated numerical limits, thereby allowing BWR plants to adopt the PTLR option. The BWROG PTLR methodology does not include development or licensing of specific RPV neutron fluence methods, but it specifies the use of NRC-approved neutron fluence methods that are consistent with RG 1.190. Further, the Hope Creek PTLR states that the RPV neutron fluence values utilized in the development of the Hope Creek P-T limit curves were calculated in accordance with RG 1.190 methods, using the NRC-approved Radiation Analysis Modeling Application (RAMA) methodology.

3.2 NRC Staff Evaluation

According to GL 96-03, there are three separate licensee actions needed in order to relocate P-T limit curves to a licensee-controlled PTLR that is governed by TS requirements. The licensee must (1) address the use of a PTLR methodology that is approved by the NRC to reference in its TS administrative controls; (2) develop the plant-specific PTLR, which contains the figures, values, parameters, and any explanation necessary; and (3) modify the applicable sections of the TSs accordingly. The NRC staff reviewed the PSEG's LAR to implement a PTLR based on the criteria of GL 96-03, as augmented by TSTF-419-A. The NRC also reviewed specific changes to the actual P-T limits based on the licensee's implementation of the latest NRC-approved BWROG PTLR methodology, as well as the licensee's incorporation of recent RPV material surveillance program data for Hope Creek. The NRC staff's findings are provided in Sections 3.2.1 through 3.2.6 of this SE.

Relocation of the P-T limit curves from the TS LCO to the PTLR does not eliminate the requirement to operate the RCS in accordance with P-T limits that are in compliance with 10 CFR Part 50, Appendix G. The requirement to operate the RCS within the limits contained in the PTLR is still specified in the P-T LCO, and the content of PTLR is controlled by the TSs. Only the actual P-T limit figures, values, and parameters associated with the P-T limits are to be relocated to the PTLR. In order for the P-T limits and associated parameters to be relocated to a PTLR, a methodology for their development must be approved in advance by the NRC, based on the guidance of GL 96-03. This NRC-approved PTLR methodology must be directly referenced in the new TS administrative controls for the PTLR.

3.2.1 TS Changes – Consistency with GL 96-03 and TSTF-419-A

The NRC staff reviewed the licensee's proposed TS revisions for implementation of the proposed PTLR to determine whether the revisions address the criteria of GL 96-03 and TSTF-419-A.

1. The NRC staff verified that the licensee's revision to TS Section 1.1, "Definitions," to include the new definition of the PTLR, "Pressure and Temperature Limits Report (PTLR)," is consistent with TSTF-419-A. Therefore, it is acceptable.

2. TS Section 3/4.4.6, "Pressure/Temperature Limits," LCO, action statements, and SRs are revised to replace all specified P-T limit figures, heatup and cooldown rates, and minimum temperature criteria with a reference to the applicable limits in the PTLR. TS Section 3/4.4.6 specifically requires RCS operation within the limits specified in the PTLR. The NRC staff verified that this is consistent with GL 96-03 and TSTF-419-A. Therefore, it is acceptable.

3. A new TS requirement has been added to the Hope Creek TS administrative controls. Specifically, Section 6.9.1.10, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," has been added. This TS section contains the TS administrative controls governing the content, methodology, and NRC reporting requirements for updates to the PTLR. The NRC staff identified that the new TS Section 6.9.1.10 does the following:

   (a) It identifies that the P-T limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, including the heatup and cooldown rates, shall be established in the PTLR.

(b) It identifies the TS sections (TS Section 3/4.4.9, LCO, action statements, and SRs) that require operation in accordance with the limits in the PTLR, per GL 96-03 and TSTF-419-A.

(c) It specifies that the analytical methods used to determine the P-T limits shall be those previously reviewed and approved by the NRC, specifically those described in BWROG-TP-11-022-A, Revision 1.

(d) It specifies that the PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

The NRC staff verified that all of the above PTLR administrative controls to be included in the new TS Section 6.9.1.10 are consistent with GL 96-03 and TSTF-419-A. Therefore, the NRC staff determined that the addition of TS Section 6.9.1.10 is acceptable.

4. Editorial revisions are made to the TS Index. The NRC staff confirmed that these editorial revisions to the index are consistent with the TS changes evaluated above for implementation of the PTLR. Therefore, they are acceptable.

If PSEG's LAR is approved by the NRC, the content of the PTLR shall be maintained in accordance with the new TS administrative controls, as identified above. Any PTLR changes that deviate from the underlying NRC-approved PTLR methodology, as referenced in the TS administrative controls, would require a new LAR in order to change the underlying TS PTLR methodology. Such TS changes are required to be submitted to the NRC for review and approval as a new LAR pursuant to 10 CFR 50.90.

3.2.2 PTLR Acceptability

The NRC staff examined the proposed PTLR and determined that it was developed from the template PTLR found in Appendix B of BWROG-TP-11-022-A, Revision 1. Furthermore, the NRC staff reviewed the licensee's proposed PTLR to determine whether it satisfies the seven plant-specific technical criteria for PTLRs specified in Attachment 1 of GL 96-03. The NRC staff's review findings regarding the consistency of the proposed PTLR with the seven technical criteria for PTLRs from Attachment 1 of GL 96-03 are summarized below:

(1) Criterion 1 specifies that the PTLR should provide the values of neutron fluence that are used in the adjusted RT_{NDT} (ART) calculation for the RPV beltline materials. The NRC staff confirmed that the 32, 44, and 56 EFPY neutron fluence values for the RPV beltline materials are provided in Tables 10, 11, and 12, respectively, of the Hope Creek PTLR. Therefore, the NRC staff determined that PTLR Criterion 1 is satisfied.

(2) Criterion 2 specifies that the PTLR should provide the surveillance capsule withdrawal schedule, or reference, by title and number, the documents in which the schedule is located. Additionally, the PTLR must reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data. The NRC staff determined that the surveillance capsule withdrawal schedule is correctly identified in Appendix A of the Hope Creek PTLR and is based on the licensee's participation in the NRC staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP). The licensee also stated Hope Creek is currently committed to participating in the BWRVIP ISP during the period of extended operation. The NRC staff identified that the appropriate references for the BWRVIP ISP governing documents, including the BWRVIP ISP implementation plan and the BWRVIP ISP

evaluations of the participating plant's surveillance data, are provided in Section 6.0 of the Hope Creek PTLR. In particular, the NRC staff determined that the referencing of the ISP evaluations of the plant's surveillance data satisfies the criterion that the PTLR must reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data. Therefore, the NRC staff determined that PTLR Criterion 2 is satisfied.

(3) Criterion 3 specifies that the PTLR should provide the low-temperature overpressure protection (LTOP) system setpoint curves or parameters if LTOP system limits are relocated to the PTLR. The NRC staff noted that LTOP systems are not used for BWRs. Therefore, the NRC staff determined that this criterion is not applicable to Hope Creek.

(4) Criterion 4 specifies that the PTLR should identify the limiting ART values and limiting RPV beltline materials at the 1/4T and 3/4T locations. The NRC staff confirmed that 32, 44, and 56 EFPY ART values for all RPV beltline materials, including the limiting RPV beltline materials, are provided in Tables 10, 11, and 12 of the Hope Creek PTLR, respectively. Therefore, the NRC staff determined that PTLR Criterion 4 is satisfied. The NRC staff's review of the licensee's detailed ART calculation for the RPV beltline materials is documented below in Section 3.2.4 of this SE.

(5) Criterion 5 specifies that the PTLR should provide the P-T limit curves for heatup, cooldown, criticality, and pressure testing conditions. The NRC staff confirmed that P-T limit curves for these conditions are provided in Figures 1, 2, and 3 of the Hope Creek PTLR for 32 EFPY; Figures 4, 5, and 6 of the Hope Creek PTLR for 44 EFPY; and Figures 7, 8, and 9 of the Hope Creek PTLR for 56 EFPY. The corresponding tabulated P-T limit values for these conditions are provided in Tables 1, 2, and 3 of the Hope Creek PTLR for 32 EFPY; Tables 4, 5, and 6 of the Hope Creek PTLR for 44 EFPY; and Tables 7, 8, and 9 of the Hope Creek PTLR for 56 EFPY. Therefore, the NRC staff determined that PTLR Criterion 5 is satisfied. Since the P-T limit curves that will be implemented and placed into the PTLR are different than the curves in the current TSs, the NRC staff also performed a detailed review of the new P-T limits to determine whether they are in compliance with 10 CFR Part 50, Appendix G. The NRC staff's review of the licensee's new P-T limits is documented below in Section 3.2.5.

(6) Criterion 6 specifies that the PTLR should identify the minimum temperatures on the P-T limit curves, such as the minimum boltup temperature and the minimum hydrostatic test temperature. The staff confirmed that the applicable minimum temperature criteria, including the minimum boltup temperature, minimum temperature for pressure greater than 20 percent of the preservice hydrostatic test pressure, and the minimum temperature for criticality are identified in the P-T limit curves provided in Figures 1, 2, and 3 of the Hope Creek PTLR for 32 EFPY; Figures 4, 5, and 6 of the Hope Creek PTLR for 44 EFPY; and Figures 7, 8, and 9 of the Hope Creek PTLR for 56 EFPY. Therefore, the NRC staff determined that PTLR Criterion 6 is satisfied.

(7) Criterion 7 specifies that the PTLR should provide RPV surveillance data and calculations of the CF in the PTLR if the RPV surveillance data are used in the ART calculation. Criterion 7 also specifies that the PTLR should evaluate the RPV surveillance data to determine if it meets the credibility criteria of RG 1.99, Revision 2, and provide the results of the credibility assessment. The NRC staff noted that Tables 10, 11, and 12 of the proposed Hope Creek PTLR list several CF values that are

based on RPV surveillance data from the BWRVIP ISP, including those for the limiting RPV beltline materials.

In its September 5, 2017, supplement, the licensee provided ISP data and calculations from the BWRVIP-135 report that were used to determine the Hope Creek ISP CF and ART values listed in Tables 10, 11, and 12 of the proposed Hope Creek PTLR. The NRC staff reviewed this ISP data and confirmed that it was correctly applied for determining these CF and ART values. The NRC staff also confirmed that the credibility of the ISP data was correctly evaluated based on the criteria of RG 1.99, Revision 2. Therefore, the NRC staff determined that PTLR Criterion 7 is satisfied.

Based on the information above, the NRC staff determined that the licensee has met the seven criteria set forth in GL 96-03, and that the Hope Creek PTLR is acceptable for incorporation by reference in TS Sections 3/4.4.6 and 6.9.1.10.

3.2.3 Consideration of Surveillance Program Data

The most recent BWRVIP ISP surveillance capsule was pulled from Hope Creek's RPV in 2015. The surveillance capsule test report was submitted to the NRC from Electric Power Research Institute by letter dated October 31, 2016 (ADAMS Accession No. ML17025A035), in compliance with the requirements of 10 CFR Part 50, Appendix H. The NRC staff identified that the surveillance weld material from this capsule was a heat-to-heat match with several of the Hope Creek RPV beltline welds, and surveillance weld test data showed that the 32 EFPY P-T limits currently implemented in the TS LCOs were non-conservative. Therefore, for the subject PTLR LAR, the licensee recalculated its P-T curves based on incorporation of the new surveillance weld data. The NRC staff confirmed that this recent BWRVIP ISP surveillance weld data now defines the limiting ART value for the RPV beltline region. The NRC staff identified that the recalculated ART values and P-T curves based on the surveillance data are included in the proposed PTLR for ensuring compliance with the requirements of 10 CFR Part 50, Appendix G. The details of the ART and P-T limit curve calculations are discussed in Section 3.2.4 of this SE.

3.2.4 RPV Beltline Materials ART

RPV beltline material ART values are used in the development of the P-T limits, and the guidance for calculating these values is provided in RG 1.99, Revision 2.

In Section 5.0 of the proposed Hope Creek PTLR, the licensee described the ART calculations for all RPV beltline and extended beltline materials. The extended beltline materials are those materials with projected neutron fluence values greater than 1 x 10^{17} neutrons per square centimeter (n/cm^2), at energies (E) greater than 1 million electron volts (E > 1 mega electron volts (MeV)). The ART calculations are listed in Tables 10, 11, and 12 of the PTLR for 32, 44, and 56 EFPY respectively. The limiting ART values are listed as 90.3 degrees Fahrenheit (°F) at 32 EFPY, 103.3 °F at 44 EFPY, and 113.8 °F at 56 EFPY, corresponding to Hope Creek RPV surveillance weld heat number D53040. The licensee applied the maximum tensile stress for both the heatup and cooldown conditions at 1/4T for calculating its P-T limits. The licensee identified that this is conservative because the 1/4T material toughness is lower than the 3/4T location based on neutron embrittlement. Therefore, the licensee only provided RPV beltline ART calculations at the 1/4T location.

The NRC staff performed independent confirmatory calculations for the limiting beltline ART values at 32, 44, and 56 EFPY using the methods in RG 1.99, Revision 2. In its September 5, 2017, letter, the licensee provided additional data from the BWRVIP ISP necessary for the staff to complete the ART assessment. The NRC staff's resulting 1/4T ART values were consistent with the values calculated by the licensee. The NRC staff also confirmed that a plant-specific evaluation of ARTs at the 3/4T location is not needed, based on the fact that the BWROG-TP-11-022-A, Revision 1, methodology applies the maximum thermal tensile stress for both heatup and cooldown at the 1/4T location, and RPV beltline material neutron embrittlement at the 1/4T location is significantly greater than the 3/4T location. Based on its confirmatory calculation, the NRC staff determined that the licensee's new RPV beltline ART values are acceptable for implementation in the PTLR.

3.2.5 P-T Limit Confirmatory Calculations

The proposed PTLR contains new P-T limit curves for hydrostatic pressure and leak testing, normal operation – core not critical, and normal operation – core critical for 32, 44, and 56 EFPY. The new proposed P-T limit curves to be implemented are different than the curves in the current TSs due to the required changes resulting from the shift in the limiting RPV beltline P-T curve based on the incorporation of the most recent surveillance capsule data and other changes to P-T limits for RPV beltline and non-beltline discontinuity regions that result from the implementation of the latest NRC-approved PTLR methodology in BWROG-TP-11-022-A, Revision 1.

The PTLR methodology specifies that separate P-T limit curves be generated for the upper vessel region (including the feedwater nozzle), the RPV beltline region (including the instrumentation nozzles), and the RPV bottom head penetrations region. The NRC staff performed confirmatory calculations for each of these regions to ensure that the proposed P-T limits are at least as conservative as those determined using the methodology of the ASME Code, Section XI, Appendix G, for the specified operating conditions. The NRC staff's confirmatory calculations were performed using the methodology of the ASME Code, Section XI, Appendix G; BWROG-TP-11-022-A, Revision 1, and the ART values reported in the PTLR for the RPV beltline materials, as confirmed above. The NRC staff's confirmatory calculation also addressed the licensee's incorporation of the RPV beltline instrumentation nozzles into the proposed P-T limit curves, consistent with the NRC staff's guidance in RIS 2014-11 and BWROG-TP-11-022-A, Revision 1.

Based on its confirmatory calculations, the NRC staff verified that the licensee's proposed P-T limit curves are at least as conservative as those generated using the methods of the ASME Code, Section XI, Appendix G, as required by 10 CFR Part 50, Appendix G. The NRC staff also verified that the proposed P-T limits meet the minimum temperature requirements specified in Table 1 of 10 CFR Part 50, Appendix G, based on the RT_{NDT} for the most limiting material in the RPV closure head flange and vessel flange regions that are highly stressed by the RPV closure bolt preload. The NRC staff's confirmatory calculations also verified that the licensee's P-T limit curves were generated consistent with NRC-approved PTLR methodology in BWROG-TP-11-022-A, Revision 1. Therefore, the NRC staff finds that the licensee's revised P-T limits are in compliance with 10 CFR Part 50, Appendix G, and are acceptable. Based on its confirmatory calculations, the NRC staff also finds that there is reasonable assurance that the licensee's plant-specific implementation of the BWROG PTLR methodology, as referenced in the proposed TS administrative controls, will continue to generate future P-T limits that are in compliance with 10 CFR Part 50, Appendix G.

3.2.6 Neutron Fluence Methodology

As noted in Section 2.4 of this SE, the licensee proposed to implement the guidance provided in BWROG-TP-11-022-A, Revision 1. Regarding reactor vessel neutron fluence, Table 1-1 of the BWROG PTLR methodology states, "Fluence methods and results must comply with RG 1.190 and have NRC approval for use with this LTR." On page 4 of Attachment 1 to the LAR, the licensee indicated that the fluence calculations were performed using the RAMA fluence methodology. In its supplemental letter dated April 28, 2017, the licensee submitted document EPR-HC1-001-R-002, "Hope Creek Nuclear Generating Station Unit 1 Reactor Pressure Vessel Fluence Evaluation at End of Cycle 19 with Projections to 56 EFPY." The NRC staff evaluated this document and determined that it demonstrates that the fluence evaluation supporting the proposed PTLR is adherent to the guidance contained in RG 1.190, and is acceptable as discussed in the following sections.

3.2.6.1 Summary of Calculation

The calculation was performed in accordance with the NRC-approved RAMA fluence methodology, which is documented in BWRVIP-114NP-A, "RAMA Fluence Methodology Theory Manual, 1019049, Final Report," June 2009 (ADAMS Accession No. ML092650376). The vessel geometry was represented in three dimensions using a combination of nominal and as-built dimensions. The fuel geometry for peripheral assemblies was modeled with sufficient detail to allow a pin-wise representation of the bundle design. The method relies on S16 angular quadrature to perform the transport calculations, and nuclear cross sections are obtained from the VITAMIN-B6 and BUGLE-96[1] collapsed cross-section libraries.

The calculation provides estimated fluence results for shell plates, shell welds, and nozzle welds within the RPV beltline. The beltline is considered as the region comprising components having a total fluence greater than 1×10^{17} n/cm^2 at E > 1 MeV. The peak fluence values are provided in the report, in Chapter 2, for end-of-cycle 19 and projected to the end of the renewed operating license, which is expressed as 56 EFPY of exposure.

RIS 2014-11 states, "Specifically, all ferritic components within the entire reactor vessel must be considered in the development of P-T limits, and the effects of neutron radiation must be considered for any locations that are predicted to experience a neutron fluence exposure greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the licensed operating period." In contrast, the NRC staff SE approving the RAMA suite of topical reports[2] indicates that "[t]he reactor beltline region [is] defined by the top and bottom planes of the active fuel and the inside surface of the biological shield." Despite this difference in the extent of the beltline region, it is noted that some of the nozzle weld locations extend slightly above the active fuel region of the core. However, the NRC staff also notes that RAMA was further qualified using scrapings from reactor vessel internal components including top guide samples, which are also located above the top of active fuel plane. These conclusions appear in BWRVIP-145NP-A, "Evaluation of Susquehanna Unit 2 Top Guide and Core Shroud Material Samples Using RAMA Fluence Methodology" (ADAMS Accession No. ML100260948). These comparisons were generically accepted by the NRC staff, and for the purposes of the present evaluation, did not indicate that RAMA would produce invalid estimates of fluence at the nozzle locations, which are only slightly above and below the active fuel regions.

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[1] VITAMIN-B6 and BUGLE-96 are cross-section library names and are not formally defined acronyms.

[2] This SE appears as front matter in, among other BWRVIP topical reports, the aforementioned BWRVIP-114NP-A.

3.2.6.2 Geometrical and Material Input Data

Regulatory Position 1.1.1 of RG 1.190 recommends that the calculation modeling should be based on documented and verified plant-specific data. The Hope Creek EPR-HC1-001-R-002 report, Section 3.2, indicates that design inputs included both nominal and as-built design dimensional data, referencing both plant design information supplied to the vendor from the licensee, and surveillance capsule data from General Electric, the fabricator of the surveillance capsules. The information contained in the report indicates that the fluence model is specific to Hope Creek, and as such, the NRC staff concluded that it is consistent with the guidance contained in RG 1.190 and, therefore, is acceptable.

3.2.6.3 Core Neutron Source

According to Regulatory Position 1.2 of RG 1.190, the core neutron source should account for local fuel isotopics and, where appropriate, the effects of moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.

The data pertinent to modeling of the Hope Creek core neutron source is discussed in Section 3.4 of the Hope Creek EPR-HC1-001-R-002 report. Detailed descriptions of the core loading, including power history data from multiple state points (typically 10 – 20) per cycle, are included. Specific state point data were obtained from the core simulator. The flux projection for cycles beyond Cycle 19 was based on Cycle 18, as the report notes that Cycle 19 operated in an unusually rodded configuration. This configuration caused a distortion of the peripheral power shaping relative to other operating cycles. The NRC staff verified that the Cycle 18 cycle exposure at 1.4 EFPY was consistent with all other extended power uprate operating cycles, which indicated that the use of Cycle 18 data for a forward projection is reasonable.

The use of the detailed core history data summarized above indicates that the licensee has modeled the core neutron source adequately, in line with the guidance contained in RG 1.190. On this basis, the NRC staff determined that the core neutron source modeling is acceptable.

3.2.6.4 Transport Calculation

Regulatory Position 1.1 of RG 1.190 provides guidance for performing acceptable transport calculations. The nuclear data should be based on the most recent version of the Evaluated Nuclear Data File/Brookhaven (ENDF/B), which at the time of publication of RG 1.190 was ENDF/B-VI. Transport calculations should be carried out using P_3 angular decomposition of the scattering cross sections. Finally, minimum S_8 angular quadrature should also be used in the transport calculations.

The RAMA methodology adheres to this guidance, and the nuclear data is based on ENDF/B-VI. The inelastic scattering cross-section approximation is P_7 for all nuclides, with the exception of actinides and zirconium for which P_5 expansion is used. The quadrature approximation is S_8. Thus, the RAMA calculations adhere to the guidance contained in RG 1.190 and are, therefore, acceptable.

3.2.6.5 Qualification

The qualification guidance contained in Regulatory Position 1.4 of RG 1.190 is summarized as follows:

> The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.

In addition, Regulatory Position 1.4 of RG 1.190 recommends a 20-percent allowance for the uncertainty at the 1-sigma level and suggests that agreement between measured and calculated fluence values for the qualification exercises be within plus or minus 20 percent. Adherence to this guidance ensures that the margins provided for fluence in the temperature shift calculations required by 10 CFR Part 50, Appendix G, are bounding of the uncertainties associated with the calculated, best-estimate fluence values.

The generic qualification of the RAMA fluence methodology against standard experimental and calculational benchmarks recommended by RG 1.190 is documented in BWRVIP-115-A, "RAMA Fluence Methodology Benchmarks Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems" (ADAMS Accession No. ML100540367). The NRC staff's SE approving the RAMA methodology acknowledges that the benchmarking is adequate for the purposes of adherence to RG 1.190, and on that basis, the NRC staff determined that the methodology is suitably qualified regarding comparison to measurement and calculational benchmarks.

In addition to the standard exercises addressed in BWRVIP-115-A, the NRC staff also determined that the RAMA methodology is acceptably qualified with plant-specific comparisons to operating reactor dosimetry. Supporting the BWR/4 vessel geometry specifically, the NRC staff previously considered comparisons to dosimetry at both Hope Creek and Susquehanna Steam Electric Station. The NRC staff reviewed these comparisons as a part of the generic review of RAMA and determined that they demonstrate adequate agreement between RAMA-calculated fluence values and the exposure measured for the specimens contained in the dosimetry capsules. Since Hope Creek was included in this qualification effort, the NRC staff concludes that the generic qualification of RAMA is applicable to Hope Creek.

The calculation supporting the Hope Creek PTLR application included additional comparisons to Hope Creek surveillance capsule dosimetry. Specifically, comparisons were provided for surveillance capsules withdrawn following Cycles 5 and 19. Using as-built dimensions, the ratio of calculated-to-measured fluence values for specimens in both capsules was generally adequate (i.e., within 20 percent with the exception of a single nickel dosimeter), but the use of nominal dimensions produced better agreement. Given the existence of a dosimetry comparison that exceeded the 20-percent allowance, the NRC staff reviewed the capsule dosimetry comparison in greater detail.

The discussion in the introduction of Chapter 5 of the Hope Creek EPR-HC1-001-R-002 report notes that an as-built dimension for the capsule mounting bracket differed from the nominal dimension by an amount that exceeded the estimated tolerance for the dimension. The report notes that a cause for this discrepancy may be that the as-built dimension is reported relative to the RPV base metal, while the nominal dimension is reported relative to the RPV cladding inner surface. The calculation also notes that the flux wire comparison from Cycle 1 was unaffected because the capsule was effectively in contact with the RPV clad inner surface. The NRC staff reviewed the dosimetry comparison results and determined that they are acceptable based on the following considerations: (1) even considering the as-built dimensions, the general agreement between measured and calculated values is within RG 1.190 allowances; (2) the use of the nominal dimensions produced improved agreement; (3) the qualitative investigation produced reasonable evidence that the discrepancy in Cycles 5 and 19 capsule comparisons was associated with hardware specific to the capsules themselves, meaning the vessel fluence projections would be unaffected; and (4) notwithstanding the Hope Creek specific comparisons, the qualification of the RAMA methodology for BWR/4 geometry has been generically accepted.

The analytic uncertainty analysis methods are discussed in Chapter 7.3 of BWRVIP-114-A. These methods are applied to the calculation provided in Chapter 6 of the Hope Creek EPR-HC1-001-R-002 report with an estimated combined uncertainty for fluence of 9.9 percent. This is well within the 20-percent allowance provided in RG 1.190, and is, therefore, acceptable.

3.2.6.6 Fluence Conclusion

As discussed above, the NRC staff review confirmed that the RAMA methodology is NRC-approved and RG 1.190-adherent, and that the methodology has been acceptably qualified for use and applied to Hope Creek. Based on these considerations, the NRC staff determined that the fluence calculations supporting the proposed PTLR implementation are acceptable.

3.3 Technical Conclusion

Based on its evaluation as documented in Section 3.2 of this SE, the NRC staff has determined that the proposed TS changes are consistent with the criteria of GL 96-03 and TSTF-419-A. The NRC staff has also determined that the proposed PTLR is consistent with the seven PTLR technical criteria identified in GL 96-03; therefore, it is acceptable for implementation in accordance with the requirements of TS Section 6.9.1.10, as requested in this amendment. Finally, the NRC staff has determined that the new P-T limits and associated parameters contained in the proposed PTLR satisfy the requirements of 10 CFR Part 50, Appendix G, and are also consistent with the latest NRC-approved BWR PTLR methodology documented in BWROG-TP-11-022-A, Revision 1. Therefore, the NRC staff concludes that the licensee's proposed TS changes and accompanying PTLR are acceptable for implementation at Hope Creek.

Upon implementation of this amendment, future changes to the content of the PTLR are to be administratively controlled in accordance with the requirements of TS Section 6.9.1.10, as established in this amendment. The PTLR shall be provided to the NRC upon issuance for each RPV neutron fluence period, and for any revision or supplement thereto, in accordance with paragraph "c" of TS Section 6.9.1.10. Any change to the PTLR that deviates from the BWROG-TP-11-022-A, Revision 1, methodology, as specified in TS Section 6.9.1.10, shall require the submittal of a new LAR in order to change the TSs pursuant to 10 CFR 50.90.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment on December 4, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (82 FR 23628; May 23, 2017). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: Benjamin Parks
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Date: December 14, 2017

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT TO REVISE AND RELOCATE THE PRESSURE-TEMPERATURE LIMIT CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT (CAC NO. MF9502; EPID L-2017-LLA-0204) DATED DECEMBER 14, 2017

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