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

November 1, 2021

Mr. John J. Grabnar
Site Vice President
Energy Harbor Nuclear Corp.
Beaver Valley Power Station
Mail Stop P-BV-SSB
P.O. Box 4, Route 168
Shippingport, PA 15077-0004

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 313 AND 203 RE: REACTOR COOLANT SYSTEM, PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2020-LLA-0233)

Dear Mr. Grabnar:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 313 and 203 to Renewed Facility Operating License Nos. DPR-66 and NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2, respectively. These amendments update the methods used to determine reactor coolant system pressure and temperature limits for operation of the Beaver Valley Power Station, Unit Nos. 1 and 2, as requested by your application dated October 30, 2020, and supplemented by letters dated April 22, 2021, and August 13, 2021.

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/

Sujata Goetz, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 313 to DPR-66
2. Amendment No. 203 to NPF-73
3. Safety Evaluation

cc: Listserv

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

ENERGY HARBOR NUCLEAR CORP.

ENERGY HARBOR NUCLEAR GENERATION LLC

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 313

Renewed License No. DPR-66

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

   A. The application for amendment by Energy Harbor Nuclear Corp.* acting on its own behalf and as agent for Energy Harbor Nuclear Generation LLC (the licensees), dated October 30, 2020, as supplemented by letter dated April 22, 2021, and August 13, 2021, 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.

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* Energy Harbor Nuclear Corp. is authorized to act as agent for Energy Harbor Nuclear Generation LLC and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-66 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 313, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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Date: 2021.11.01 16:09:06 -04'00'

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

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

Date of Issuance: November 1, 2021

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

ENERGY HARBOR NUCLEAR CORP.

ENERGY HARBOR NUCLEAR GENERATION LLC

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 203

Renewed License No. NPF-73

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

   A. The application for amendment by Energy Harbor Nuclear Corp., acting on its own behalf and as agent for Energy Harbor Nuclear Generation LLC* (the licensees), dated October 30, 2020, as supplemented by letter dated April 22, 2021, and August 13, 2021, 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.

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* Energy Harbor Nuclear Corp. is authorized to act as agent for Energy Harbor Nuclear Generation LLC and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

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-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 203, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. Energy Harbor Nuclear Corp. 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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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: November 1, 2021

ATTACHMENT TO LICENSE AMENDMENT NOS. 313 AND 203

BEAVER VALLEY POWER STATION, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

DOCKET NOS. 50-334 AND 50-412

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Renewed Facility Operating License No. DPR-66

Remove	Insert
Page 3	Page 3

Renewed Facility Operating License No. NPF-73

Remove	Insert
Page 4	Page 4

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.

Appendix A, Technical Specifications

Remove	Insert
Page 5.6 – 3	Page 5.6 – 3
Page 5.6 – 4	Page 5.6 – 4
	Page 5.6 – 5

(3) FENOC, 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) FENOC, 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;

(5) FENOC, 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.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

FENOC is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

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

The Technical Specifications contained in Appendix A, as revised through Amendment No.313, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Auxiliary River Water System</u>

(Deleted by Amendment No. 8)

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the term or conditions of any lease agreements executed as part of these transactions; (ii) the BVPS Operating Agreement, (iii) the existing property insurance coverage for BVPS Unit 2, and (iv) any action by a lessor or others that may have adverse effect on the safe operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations set forth in 10 CFR Chapter 1 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

FENOC is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.203, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON,"

WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology,"

WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™,"

WCAP-17661-P-A, "Improved RAOC and CAOC F_Q Surveillance Technical Specifications."

As described in reference documents listed above, when an initial assumed power level of 102% of RATED THERMAL POWER is specified in a previously approved method, 100.6% of RATED THERMAL POWER may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √ ™ System"

Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √ ™ System"

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, Overpressure Protection System (OPPS) enable temperature, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

LCO 3.4.12, "Overpressure Protection System (OPPS)"

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 WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

5.6 Reporting Requirements

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

WCAP-18124-NPA, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2012, may be used as an alternative to Section 2.2 of WCAP-14040-A, Revision 4.

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

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.6 Steam Generator (SG) Tube Inspection Report

5.6.6.1 Unit 1 SG Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.1, “Unit 1 SG Program.” The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service-induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.6 Reporting Requirements

5.6.6 Steam Generator (SG) Tube Inspection Report (continued)

5.6.6.2 Unit 2 SG Tube Inspection Report

1. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.2, “Unit 2 SG Program.” The report shall include:

   a. The scope of inspections performed on each SG,

   b. Degradation mechanisms found,

   c. Nondestructive examination techniques utilized for each degradation mechanism,

   d. Location, orientation (if linear), and measured sizes (if available) of service-induced indications,

   e. Number of tubes plugged or repaired during the inspection outage for each degradation mechanism,

   f. The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each steam generator,

   g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

   h. Repair method utilized and the number of tubes repaired by each repair method.

2. A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.2, “Unit 2 SG Program,” when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

3. For implementation of the voltage-based plugging or repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 313 AND 203 TO RENEWED

FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

ENERGY HARBOR NUCLEAR GENERATION LLC

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated October 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20304A215), supplemented by letters dated April 22, 2021 (ADAMS Accession No. ML21113A044), and August 13, 2021 (ADAMS Accession No. ML21228A125), Energy Harbor Nuclear Corp. (the licensee) submitted a license amendment request proposing changes to Technical Specification (TS) 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to update the methods used to determine reactor coolant system pressure and temperature (P-T) limits for operation of the Beaver Valley Power Station, Unit Nos. 1 and 2.

Currently, TS 5.6.4 requires use of WCAP-14040-A, Revision 3 (ADAMS Accession No. ML021580226), to determine RCS P-T limits. The licensee proposed changes to TSs that would allow use of WCAP-14040-A, Revision 4 (ADAMS Accession No. ML050120209), instead of WCAP-14040-A, Revision 3, for determination of P-T limits. The proposed changes would also allow the licensee to use methods described in topical report WCAP-18124-NP-A, Revision 0 (ADAMS Accession No. ML18204A010), to calculate fast (energy greater than 1 million electron volts (E > 1 MeV)) neutron fluence in ferritic components of the reactor pressure vessel (RPV) as an alternative to the fluence methods described in Section 2.2 of WCAP-14040-A, Revision 4. Fast (E > 1 MeV). Neutron fluence is used as an input when determining the RPV P-T limits.

The licensee provided additional information in response to U.S. Nuclear Regulatory Commission (NRC) staff questions in supplemental letters, dated April 22, 2021, and August 13, 2021, that clarified the application but 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 as published in Volume 85 of the *Federal Register*, page 55514, on September 8, 2020 (85 FR 55514).

2.0 REGULATORY EVALUATION

2.1 Description of RAPTOR-M3G

RAPTOR-M3G is a three-dimensional discrete ordinates radiation transport code that approximates a solution to the Boltzmann transport equation. The code methodology and qualification data are documented in the topical report WCAP-18124-NP-A, Revision 0. This topical report has been approved by NRC staff for calculation of RPV neutron fluence (ADAMS Accession No. ML18204A010), provided that limitations and conditions of the associated safety evaluation (SE) are met.

2.2 Applicable Regulatory Requirements

The NRC has established requirements in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The NRC staff evaluates the acceptability of a facility's proposed P-T limits based on the following NRC regulations and guidance:

The regulations in 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation" imposes fracture toughness and material embrittlement surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 requires, in part, that facility P-T limits for the RPV be at least as conservative as those obtained by applying the linear elastic fracture mechanics (LEFM) methodology of Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 establishes requirements for a facility's surveillance program for monitoring RPV embrittlement due to neutron irradiation.

Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity," Rev. 1 (ADAMS Accession No. ML031070438), requested that licensees submit the RPV data for their plants to the staff for review, and GL 92-01, Rev. 1, Supplement 1 (ADAMS Accession No. ML031070449), requested that licensees provide and assess data from other licensees that could affect their requirements related to facility RPV material surveillance programs.

General Design Criteria (GDCs) of Appendix A, "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50 establish requirements for the integrity of the reactor coolant pressure boundary. Specifically, GDC 14, "Reactor Coolant Pressure Boundary," GDC 30, "Quality of Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary" require, in part, the design, fabrication, and maintenance of the reactor coolant pressure boundary with adequate margin to assure that the probability of rapidly propagating failure of the boundary is minimized. In particular, GDC 31 explicitly requires consideration of the effects of irradiation on material properties as well as uncertainties in determining these effects.

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, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock” (ADAMS Accession No. ML070380185) describes acceptance criteria for determining the P-T limits for ferritic materials in the beltline of the RPV based on Appendix G to Section XI of the ASME Code methodology.

Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” (ADAMS Accession No. ML010890301) provides guidance on methods for determining RPV fluence that are acceptable to the NRC staff, in accordance with GDCs 14, 30, and 31. The guidance in RG 1.190 states that an acceptable neutron fluence calculation has the following attributes:

- Fluence estimation using an appropriate calculational methodology
- Analytic uncertainty analysis identifying possible sources of uncertainty
- Comparisons with benchmark measurements and calculations from applicable test facilities including:
    - Plant-specific operating reactor measurements
    - Pressure vessel simulator measurements
    - Calculational benchmarks

RG 1.190 is specific to neutron fluence calculations in the beltline region with close proximity to the active fuel region of the core. The licensee has performed neutron fluence calculations for both beltline and extended beltline[1] regions and the justification for use of those calculations is evaluated in Section 0 of this SE, consistent with the limitations and conditions in the NRC staff’s SE for Westinghouse Electric Company, “WCAP-18124-NP-A, Revision 0, Fluence Determination with RAPTOR-M3G and FERRET,” July 2018 (ADAMS Accession No. ML18204A010), (hereafter referred to as WCAP-18124).

RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials;” Revision 2 contains guidance for RPV embrittlement integrity evaluations.

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), clarifies that P-T limits for ferritic RPV components, such as RPV inlet and outlet nozzles, could be more limiting because higher stress levels from structural discontinuities could result in a lower allowable pressure.
RIS 2014-11 also clarifies that the RPV beltline definition in Appendix G to 10 CFR Part 50 is applicable to all RPV ferritic materials with projected fast neutron fluence values greater than 1E+17 neutrons per square centimeters ($n/cm^{2)}$ (E > 1 MeV), and that this fluence threshold remains applicable for the design life as well as throughout the licensed operating period of the reactor.

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[1] As noted in NRC Regulatory Issue Summary (RIS) 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components” (ADAMS Accession No. ML14149A165), the term “beltline” is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1×10^{17} neutrons per square centimeter (n/cm^2). In this SE, the phrase “extended beltline” is intended to refer to those beltline regions that are further away from the active fuel region of the core.

The requirements for TSs are set forth in 10 CFR 50.36, "Technical specifications." The regulation at 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.

2.3 Description of Proposed TS Change

The change proposed by the licensee would revise TS 5.6.4.b, which lists analytical methods that can be used to determine reactor coolant system pressure and temperature limits. The proposed change would add WCAP-18124-NP-A, Revision 0, to this list, as an alternative to fluence calculation methods described in Section 2.2 of WCAP-14040-A and update the methodology to WCAP-14040-A, Revision 4.

3.0 TECHNICAL EVALUATION

The staff performed an evaluation to determine whether the proposed change will continue to meet the requirements listed in Section 2.2 above.

3.1 WCAP-18124-NP-A, Revision 0

The licensee stated in Enclosure B to the LAR, that the fluence calculations were performed using RAPTOR-M3G.

The licensee stated that the neutron fluence calculations were performed in a manner consistent with the guidance described in RG 1.190. The licensee approximated a solution to the Boltzmann transport equation using the three-dimensional discrete ordinates radiation transport code RAPTOR-M3G, as described in WCAP-18124.

The licensee uses the BUGLE-96 cross-section library, which utilizes 47-neutron and 20-gamma-ray groups specifically for light water reactor applications (Radiation Safety Information Computational Center, Oak Ridge National Laboratory, "Code Package DLC-185, BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR [Light-Water Reactor] Shielding and Pressure Vessel Dosimetry Applications," July 1999).

The BUGLE-96 cross-section library is derived from the Brookhaven National Laboratory Evaluated Nuclear Data File, 6th Release (ENDF/B-VI) cross-section library. The guidance in RG 1.190 specifies that ENDF/B-VI-based nuclear data are acceptable. Anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{20} order of angular quadrature, which are both consistent with recommendations in RG 1.190. The uncertainty of RAPTOR-M3G in the core-adjacent beltline region is within the ±20 percent recommended by RG 1.190 for determination of vessel fluence.

As described above, the NRC staff reviewed the modeling approach described by the licensee and determined that the neutron fluence calculations are consistent with RG 1.190 guidance. In addition, NRC staff has approved RAPTOR-M3G and FERRET for determination of RPV fluence provided that the two limitations and conditions in the staff SE in WCAP-18124-NP-A, Revision 0, are met. The two limitations and conditions are:

- Applicability of WCAP-18124-NP, Revision 0, is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to response parameters of interest (e.g., P-T limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and reactor coolant system inlet and outlet nozzles and reactor vessel internal components.

- Least-squares adjustment is acceptable if the adjustments to the M/C (measured-to-calculated) ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the inconsistency should be disqualified.

3.1.1 Limitation and Condition 1

The licensee's RPV integrity analysis includes materials and components in an extended beltline region of the RPV. The licensee has provided justification for use of RAPTOR-M3G and FERRET in Enclosure B of the license amendment request, "Justification of Using RAPTOR-M3G for Reactor Pressure Vessel Extended Beltline Materials at Beaver Valley Units 1 and 2".

The licensee has collected measurement benchmark data through use of ex-vessel neutron dosimetry (EVND). The EVND capsules were installed at the elevation of the reactor vessel support at a 4-loop Westinghouse plant, roughly the same axial height from the core midplane as some of the major extended beltline materials analyzed in the BVPS-1 and BVPS-2 neutron fluence evaluation. The EVND capsules contain a variety of radiometric monitor foils appropriate for use in benchmarking fluence calculations. Some of the most common reactions that occur in RPV materials include those with iron, copper, and nickel. All of these materials are included in the EVND capsules and account for the lowest reaction rate uncertainties of all reactions at a 1σ uncertainty of ±5 percent. The NRC staff considers the use of EVND capsules for additional benchmarking acceptable because RG 1.190 indicates that EVND is an acceptable means of qualifying fluence estimates and because the EVND under consideration had been installed in an upper elevation where the RG indicates that cavity streaming effects may have a more dominant influence on the total fluence, in comparison to a core midplane location.

The licensee provided M/C ratios of the calculated EVND capsule reaction rates and the measured data from counting laboratories. This comparison established that the average M/C ratio is 0.78 with a standard deviation of 25.5 percent. It is noted that the iron, titanium, copper, and nickel capsules dosimeters have M/C ratios consistently below the average M/C ratio with small standard deviations. Additionally, the licensee provided best-estimate-to-calculated (BE/C) ratios for the fluence rate and iron atom displacement rate[2]. This comparison determined that the BE/C ratio for the fluence rate is 0.84 with a standard deviation of 8.9 percent and a BE/C ratio for iron atom displacement rate of 0.93 with a standard deviation of

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[2]As noted above, adjusted (best-estimate) fluence values were not used in the determination of P/T limits. However, the licensee did submit adjusted fluence values in order to provide additional validation data for the method.

11 percent. These three comparisons demonstrate that the calculations consistently over-predict the neutron exposure. Additionally, both of the BE/C ratios are within the 20 percent limit described in regulatory position 1.4.2, “Comparisons with Benchmark Methods and Calculations”, of RG 1.190. Regulatory position 1.4.2 of RG 1.190 also describes an acceptable 30 percent limit for cavity dosimetry, which the NRC staff considered in the justification for the average M/C reaction rate ratio of 0.78.

Guidance in RG 1.190 describes an acceptable uncertainty in fast neutron fluence calculations of within ±20 percent. The analytic uncertainty described in WCAP-18124 is about ±19-20 percent at the top and bottom of the active fuel. This is consistent with the guidance in RG 1.190. According to the licensee’s evaluation included in Enclosure B in the LAR, higher levels of uncertainty are estimated for the extended beltline region. However, RG 1.190 suggests that more approximate methods for determining the fluence may be appropriate when there is a large margin to the reference temperature for nil-ductility transition (RT_{NDT}) limits. The benchmarking discussed above indicates that RAPTOR-M3G consistently overpredicts the fast neutron fluence. The licensee also noted that none of the extended beltline materials in Beaver Valley Unit 1 are limiting and have significant margin to become limiting.

The NRC staff reviewed the extended beltline materials, and their associated adjusted reference temperature at ¼ thickness, and determined that the most limiting extended beltline material for Unit 1 is the upper to intermediate shell girth weld. The NRC staff performed a sensitivity study on the uncertainty required for the weld to become the most limiting RPV material. The results of that study show that a departure from the estimated fluence of 360 percent would be required for the upper to intermediate shell weld to be the most limiting material. Based on a review of the benchmarking data, a departure of 360 percent is equivalent to roughly 12σ, and is therefore a significant difference from the uncertainty estimated by the licensee.

The licensee has also estimated levels of uncertainty higher than ±20 percent in the Beaver Valley Unit 2 extended beltline. However, unlike Unit 1, some of these materials may be limiting with respect to those in the traditional beltline. Although these regions accumulate less fluence than regions closer to the active fuel, they may become limiting due to higher unirradiated RT_{NDT} values and chemical compositions that make them more sensitive to embrittlement.

The licensee specifically noted that the Unit 2 nozzle shell, nozzle shell longitudinal welds, or nozzle-to-intermediate shell weld may be limiting with respect to the traditional beltline materials. The licensee indicated in its letter dated April 22, 2021, that the analytic uncertainty of fast neutron (E > 1 MeV) fluence for these regions is estimated to be approximately ±25 percent. According to the licensee, the analytic uncertainty estimate accounted for a variety of factors considered significant to fluence in the extended beltline, including order of angular quadrature and material mixture modeling of upper and lower RPV internals.

RG 1.190 states that when calculational uncertainty is greater than ±20 percent, the model must be adjusted or a correction applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20 percent. Equation 6 of RG 1.190 can be used to calculate a correction factor for this purpose. Although the licensee has estimated analytical uncertainty to be approximately ±25 percent in this region, it conservatively proposed to assume ±30 percent uncertainty, resulting in a correction factor of 1.1 according to equation 6 of RG 1.190. Because this factor is conservative with respect to the analytic uncertainty and benchmarking data, and because the analytic uncertainty estimation appears to have accounted for factors thought to be important to the fluence estimate in the extended beltline, the NRC staff finds this treatment acceptable.

As described in "NRC Regulatory Issue Summary 2014-11: Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," October 14, 2014 (ADAMS Accession No. ML14149A165), structural discontinuities in the RPV (like nozzles) can induce additional stresses to those experienced in the traditional beltline. Therefore, additional justification is required to conclude that RPV nozzles are non-limiting with respect to beltline materials.

To address this, the licensee cited Pressurized Water Reactor Owners Group (PWROG) Report, "PWROG-15109-NP-A, Revision 0, PWR Pressure Vessel Nozzle Appendix G Evaluation," January 2020 (ADAMS Accession No. ML20024E573) which, as modified by its SE, allows embrittlement effects with respect to fracture toughness requirements for inlet and outlet nozzles of PWR RPVs to be neglected if the nozzle fast neutron (E > 1 MeV) fluence is less than 4.28×10^{17} n/cm^2. The licensee stated in its application, that estimated nozzle fast neutron (E > 1 MeV) fluence values for both Unit 1 and Unit 2 are less than half of this threshold.

The licensee evaluated the uncertainty of the nozzle fluence to be approximately ±42 percent or less. Based on this margin and uncertainty, the licensee asserts that it is not credible for the inlet and outlet nozzle fluence to exceed this threshold within the projected 54 effective full power years (EFPY) license extension period. The licensee's analytic uncertainty estimation incorporates factors specifically listed in RG 1.190 for their importance to beltline fluence analysis (such as material and geometrical representation of core and vessel internals, the neutron source, and numerical schemes used in the calculation), as well as other factors considered important to extended beltline fluence, such as quadrature order and material mixture modeling of upper and lower RPV internals.

The NRC staff reviewed differences in the way that biological shield components are modeled between the two units, since previous studies have noted sensitivity of fluence estimates in RPV nozzles to approximations used in modeling the biological shield. The Unit 1 RAPTOR-M3G model is described in Westinghouse Electric Company, "WCAP-17896-NP, Revision 0 – Analysis of Capsule X from the FirstEnergy Nuclear Operating Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," September 2014 (ADAMS Accession No. ML14288A393), which is cited as the source of Unit 1 RAPTOR-M3G fluence estimates in Table 1 of the licensee's letter dated April 22, 2021. The Unit 2 RAPTOR-M3G model is described in WCAP-18559-NP, Revision 1, included as an attachment to letter dated April 22, 2021. Both models extend radially from the center of the core to a location inside the biological shield.

As described in their respective updated final safety analysis reports, the biological shield for both Unit 1 and Unit 2 consists of a water-filled shield tank surrounded by reinforced concrete. In WCAP-17896-NP, Revision 0, Figures 6-1 through 6-4 show that the shield tank wall and shield tank water are modeled as separate regions. The radial outer boundary of the model is within the shield tank. In contrast, Figures 2-2 through 2-5 of WCAP-18559-NP, Revision 1, show the outer radial boundary in the concrete portion of the shield. However, the shield is depicted as one homogeneous region, with no differentiation between the shield tank wall, shield tank water, or the concrete shield. It is unclear whether these different depictions reflect actual differences between models for the two units.

To evaluate whether these apparent differences impacted the nozzle fluence calculations, the NRC staff compared Unit 1 and Unit 2 predictions of fast (E > 1MeV) fluence and iron displacements per atom (DPA) in the extended beltline. While the NRC staff did find that estimated fast (E > 1MeV) fluence was slightly higher for Unit 1 than Unit 2, the discrepancy was not large enough to warrant concern that the Unit 2 nozzle fluence might exceed the 4.28 x 10^{17} n/cm^2 threshold, even if the difference were entirely attributable to modeling of the biological shield. Further, no discrepancy was evident in estimates of DPA between the two units. Therefore, the NRC staff concluded that either the reporting discrepancies do not reflect actual differences between the models, or that the differences do not have a significant impact on estimation of the fluence. Since the licensee established that there was sufficient margin, accounting for the estimated uncertainty, to the threshold established in PWROG-15109, the NRC staff determined that the licensee's treatment for the nozzles was acceptable.

Therefore, the NRC staff concludes that the justification for application of the RAPTOR-M3G fluence methodology in extended beltline regions is acceptable, and that the first limitation and condition in the NRC staff SE for WCAP-18124 is satisfied.

3.1.2 Limitation and Condition 2

The licensee stated that the second limitation and condition does not apply, because the least-squares procedures were not used to adjust the calculated fast neutron fluence values for RPV materials evaluated in the reactor vessel integrity analysis. Since the licensee did not use the FERRET least-squares adjustment methods in its estimation of the fluence values in the reactor vessel integrity analysis, the NRC staff determined that this limitation does not apply, and the licensee has, therefore, addressed the limitation acceptably.

3.1.3 Technical Conclusion for the use of WCAP-18124-NP-A, Revision 0

Based on the considerations discussed above, the NRC staff has determined that use of WCAP-18124-NP-A, Revision 0, to calculate neutron fluence in the RPV for the purpose of determining RCS P-T limits for BVPS-1 and BVPS-2 up to 54 EFPY is acceptable. WCAP-18124-NP-A, Revision 0, has been generically approved by the NRC staff for this purpose. The licensee has acceptably addressed all the limitations and conditions discussed in the NRC staff's SE of WCAP-18124-NP-A, Revision 0. Further, the NRC staff finds that the neutron fluence calculation is consistent with the guidance in RG 1.190 and meets the requirements of GDC 14, 30, and 31.

3.2 <u>WCAP-14040-A, Revision 4</u>

3.2.1 Technical Evaluation for WCAP-14040-A, Revision 4

The methodology contained in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (ADAMS Accession No. ML050120209) was approved for licensing applications subject to three conditions as described in the associated NRC staff SE for WCAP-14040 Revision 4. This report describes an NRC-approved methodology for developing heatup and cooldown curves, also referred to as P-T limits, which ensure the structural integrity of the reactor vessel through the imposition of limits on RCS operation. The NRC staff reviewed the information provided by the licensee regarding each of these conditions.

Condition 1

The first condition requires that the licensee provide information consistent with GL 96-03 to demonstrate that the plant maintains a reactor vessel material surveillance program in accordance with Appendix H to 10 CFR Part 50. Appendix H requires that licensees of commercial light-water nuclear power reactors with a peak neutron fluence exceeding $1x10^{17}$ n/cm2 (E > 1 MeV) at the end of the RPV design life, maintain a RPV material surveillance program that tests irradiated material specimens that are located in surveillance capsules in the RPV. Beaver Valley Unit 1 and Unit 2 exceed these neutron fluence thresholds and are subject to these requirements, and therefore must maintain RPV surveillance programs in accordance with Appendix H. Section IV.A of Appendix H currently requires that each surveillance specimen capsule withdrawal and associated test results must be the subject of a summary technical report that is to be submitted to the NRC within 18 months of the date of the capsule withdrawal (note that the regulation had previously required the submittal of this technical report within 1 year following the date of the capsule's withdrawal).

The purpose of the surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region that result from the exposure of these materials to neutron irradiation and to the thermal environment. The surveillance capsules are located in each vessel between the core and the RPV near the inner vessel wall in the beltline region. They contain material specimens that are consequently exposed to the neutron irradiation and thermal environment of the beltline region to simulate the exposure conditions of the actual vessel beltline materials

The NRC approved the current RPV surveillance capsule withdrawal schedule for Unit 1 in a letter dated July 2, 2018 (ADAMS Accession No. ML18164A082). The currently approved withdrawal schedule is consistent with the recommendations specified in ASTM Standard E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," and the end-of-life capsule withdrawal requirement in NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Section XI.M31, "Reactor Vessel Surveillance." The surveillance capsules have been withdrawn and tested in accordance with the approved Unit 1 RPV material surveillance capsule withdrawal schedule.

Beaver Valley's current RPV surveillance capsule withdrawal schedule for Unit 2 was approved by the NRC in a letter dated July 17, 2014 (ADAMS Accession No. ML13242A266). The NRC staff approved an exemption from the testing and reporting requirements of Appendix H for Unit 2 RPV surveillance Capsule Y by letter dated June 5, 2019 (ADAMS Accession No. ML19126A195); at that time, Section IV.A of Appendix H to 10 CFR Part 50 required that each surveillance specimen capsule withdrawal and associated test results must be the subject of a summary technical report that is to be submitted to the NRC within 1 year of the date of the capsule withdrawal. As stated in the exemption, Capsule Y was removed from the RPV on October 29, 2018. The content of the capsule was placed in storage so that it could be tested in the future if Unit 2 remained operational.

The exemption to the performance of the mechanical testing of Capsule Y was approved based on the expectation that Unit 2 would cease power operation by October 31, 2021. The exemption recited the licensee's statement that if a decision is made to operate Unit 2 beyond October 31, 2021, a revised capsule testing schedule would be submitted for NRC approval prior to October 31, 2021. By letter dated October 20, 2020, the summary technical report for surveillance Capsule Y testing and analysis required by Appendix H for Unit 2 was documented in WCAP-18558-NP (ADAMS Accession No. ML20302A376). Since the testing was completed and the summary technical report was submitted prior to October 31, 2021, the licensee has met its reporting requirements under Appendix H, and there is no need for the licensee to submit a revision of the surveillance capsule testing schedule for Unit 2 Capsule Y. Therefore, the staff finds that the information submitted by the licensee has met the first condition of the SE for WCAP-14040-A, Revision 4.

Condition 2

Condition 2 requires that Revision 4 of WCAP-14040-A reflect the NRC staff conclusion that no exemption is required for licensee use of provisions in ASME Code Cases N-588, N-640 or N-641 in conjunction with the basic methodology contained in WCAP-14040, Revision 3, since these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. The current P-T limits values in the Beaver Valley Unit 1 and Unit 2 PTLR (ADAMS Accession Nos. ML19105A881 and ML14133A107, respectively) were determined using methods consistent WCAP-14040-A and the plant-specific allowances to use ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T [Pressure and Temperature] Limits for Section XI, Division 1." ASME Code Case N-640 allows the use of K_{IC} (which is the material toughness property measured in terms of stress intensity factor, K_i, which will lead to nonductile crack propagation) instead of K_{IA} (which is the critical value of the stress intensity factor, K_I, for crack arrest as a function of temperature) in the development of P-T limit curves. The licensee did not utilize Code Cases N-588 and N-641, and those Code Cases are therefore not applicable for Beaver Valley.

The P-T limit methodology in the current PTLRs for Unit 1 and Unit 2 is consistent with WCAP-14040-A, Rev. 2, which had included an exception for the use of ASME Code Case N-640. WCAP-14040-A, Revision 4, has been revised to show that ASME Code Case N-640 has been incorporated by reference into 10 CFR 50.55a. Since an exemption for the use of ASME Code Case N-640 is no longer necessary, the licensee has updated the PTLR to reflect the incorporation of Code Case N-640 by reference in 10 CFR 50.55a.

The PTLR methodology used by Beaver Valley Unit 1 and Unit 2 to generate P-T limits curves based on WCAP-14040-A, Revision 4 is consistent with ASME Code Section XI, Appendix G, and continues to generate the plant-specific allowances from ASME Code Case N-640. The licensee has updated the information in the PTLR to reflect the incorporation of ASME Code Case N-640 in the ASME Code as incorporated by reference in 10 CFR 50.55a. Therefore, the NRC staff concludes that the updated information has adequately addressed the second condition in the SE to WCAP-14040-A, Revision 4 and the second condition has been met.

Condition 3

The third condition requires that the RPV flange minimum temperature requirements be incorporated into a facility's P-T curves until Appendix G to 10 CFR Part 50 is revised to modify the existing RPV flange minimum temperature requirement or an exemption request to modify these requirements is approved by the NRC for a specific facility. The licensee referred to WCAP-14040-A, Revision 4, Section 2.9 ("Closure Head/Vessel Flange Requirements") which reiterates this position. Appendix G has not been revised since WCAP-14040-A, Rev. 4, including the associated SE, was issued on February 27, 2004. Therefore, the staff concludes that the third condition in the SE to WCAP-14040-A, Revision 4, has been adequately addressed because there have been no changes to the requirement of Appendix G to 10 CFR Part 50 regarding the reactor vessel flange minimum temperature requirements, and the licensee is not requesting a plant-specific exemption to the requirements,

3.2.2 Technical Conclusion for WCAP-14040-A, Revision 4

As detailed in Section 3.2.1 of this SE, the licensee has adequately addressed the three conditions required in the SE for the use of WCAP-14040-A, Revision 4. For the first condition, additional information was provided to support the reactor vessel material surveillance program requirements as described in GL 96-03. Information regarding the most recent surveillance capsule summary technical reports were included to reflect the current status of these programs. The NRC staff found that the surveillance programs continue to meet the requirements of Appendix H to 10 CFR Part 50 to monitor changes in fracture toughness of the ferritic beltline materials through the scheduled testing of surveillance capsule materials.

For the second condition, both Unit 1 and Unit 2 P-T curves incorporated ASME Code Case N-640. WCAP-14040-A, Revision 4, has been revised to show that ASME Code Case N-640 has been incorporated by reference into 10 CFR 50.55a. The licensee has reflected these changes in the PTLR. This satisfied the second condition.

The third condition addressed RPV flange minimum temperature requirements per Appendix G to 10 CFR Part 50. Since Appendix G to 10 CFR Part 50 has not been revised, there is no modification to the approved PTLR methodology necessary at this time. The licensee further stated that in accordance with Section 2.9 of WCAP-14040-A, Revision 4, the reactor vessel flange minimum temperature requirement will continue to be incorporated into the Beaver Valley Unit 1 and Unit 2 P-T limit curves, until Appendix G to 10 CFR Part 50 is revised to modify or eliminate the existing RPV flange requirements, or an exemption request to modify or eliminate these requirements is approved by the NRC. This is consistent with the current requirements of Appendix G to 10 CFR Part 50, therefore the staff concluded that the third condition was met.

Therefore, the staff has concluded that all three conditions for amending the PTLR methodology to WCAP-14040-A, Revision 4, for Beaver Valley Unit 1 and Unit 2 have been met by the licensee.

3.3 Technical Conclusion

Based on the NRC staff's review of the information provided in the licensee's submittals, the NRC staff concludes that the proposed change to TS 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," meets the conditions necessary for updating the PTLR methodology used to determine the P-T limits in WCAP-14040-A, Revision 4. Additionally, the NRC staff concludes that the fluence calculation provided to the staff is consistent with the guidance in RG 1.190, that the limitations and conditions associated with WCAP-18124-NP-A have been met, and that use of WCAP-18124-NP-A, Revision 0, as an alternative to Section 2.2 of WCAP-14040-A, Revision 4, in determining P-T limits is consistent with GDCs 14, 30, and 31. Further, the staff finds that the regulatory requirements of 10 CFR 50.36(c)(5) will continue to be met because the TS, as revised by the proposed change, will continue to contain provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Therefore, the proposed change is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Commonwealth of Pennsylvania official was notified of the proposed issuance of the amendments on September 24, 2021. The official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or change SRs. The NRC staff has determined that these 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 these amendments involve no significant hazards consideration, as published in the *Federal Register* (85 FR 85674; December 29, 2020) and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributors: A. Rau
C. Fairbanks
M. Hamm
O.Yee

Date of Issuance: November 1, 2021

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 313 AND 203 RE: REACTOR COOLANT SYSTEM, PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2020-LLA-0233) DATED NOVEMBER 1, 2021

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