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

November 23, 2022

Mr. Robert Schuetz
Chief Executive Officer
Energy Northwest
76 North Power Plant Loop
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT NO. 268 TO REVISE TECHNICAL SPECIFICATION 3.4.11 “RCS PRESSURE AND TEMPERATURE (P/T) LIMITS” (EPID L-2021-LLA-0191)

Dear Mr. Schuetz:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 268 to Renewed Facility Operating License No. NPF-21 for the Columbia Generating Station (Columbia). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 13, 2021, as supplemented by letter dated July 11, 2022.

The amendment revises the reactor coolant system (RCS) pressure and temperature (P/T) curves in the TSs for Columbia. Specifically, the amendment replaces the existing P/T curves in figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 of TS 3.4.11, “RCS Pressure and Temperature (P/T) Limits,” with P/T curves valid for 54 effective full power years, which are based on analyses projected to the period of extended operation. The P/T curves also satisfy license renewal commitment number 54 in appendix A, table A-1 of NUREG-2123, “Safety Evaluation Report Related to the License Renewal of Columbia Generating Station.”

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 Dennis Galvin for/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 268 to NPF-21
2. Safety Evaluation

cc: Listserv

ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 268
License No. NPF-21

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

   A. The application for amendment by Energy Northwest (the licensee), dated October 13, 2021, as supplemented by letter dated July 11, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-21 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. 268 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee 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 from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

Digitally signed by Jennifer L. Dixon-Herrity
Date: 2022.11.23 10:51:05 -05'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-21 and the Technical Specifications

Date of Issuance: November 23, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 268 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-21

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

Replace the following pages of Renewed Facility Operating License No. NPF-21 and the Appendix A, Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
-4-	-4-

Technical Specification

REMOVE	INSERT
3.4.11-5	3.4.11-5
3.4.11-6	3.4.11-6
3.4.11-7	3.4.11-7

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 268 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

a. For Surveillance Requirements (SRs) not previously performed by existing SRs or other plant tests, the requirement will be considered met on the implementation date and the next required test will be at the interval specified in the Technical Specifications as revised in Amendment No. 149.

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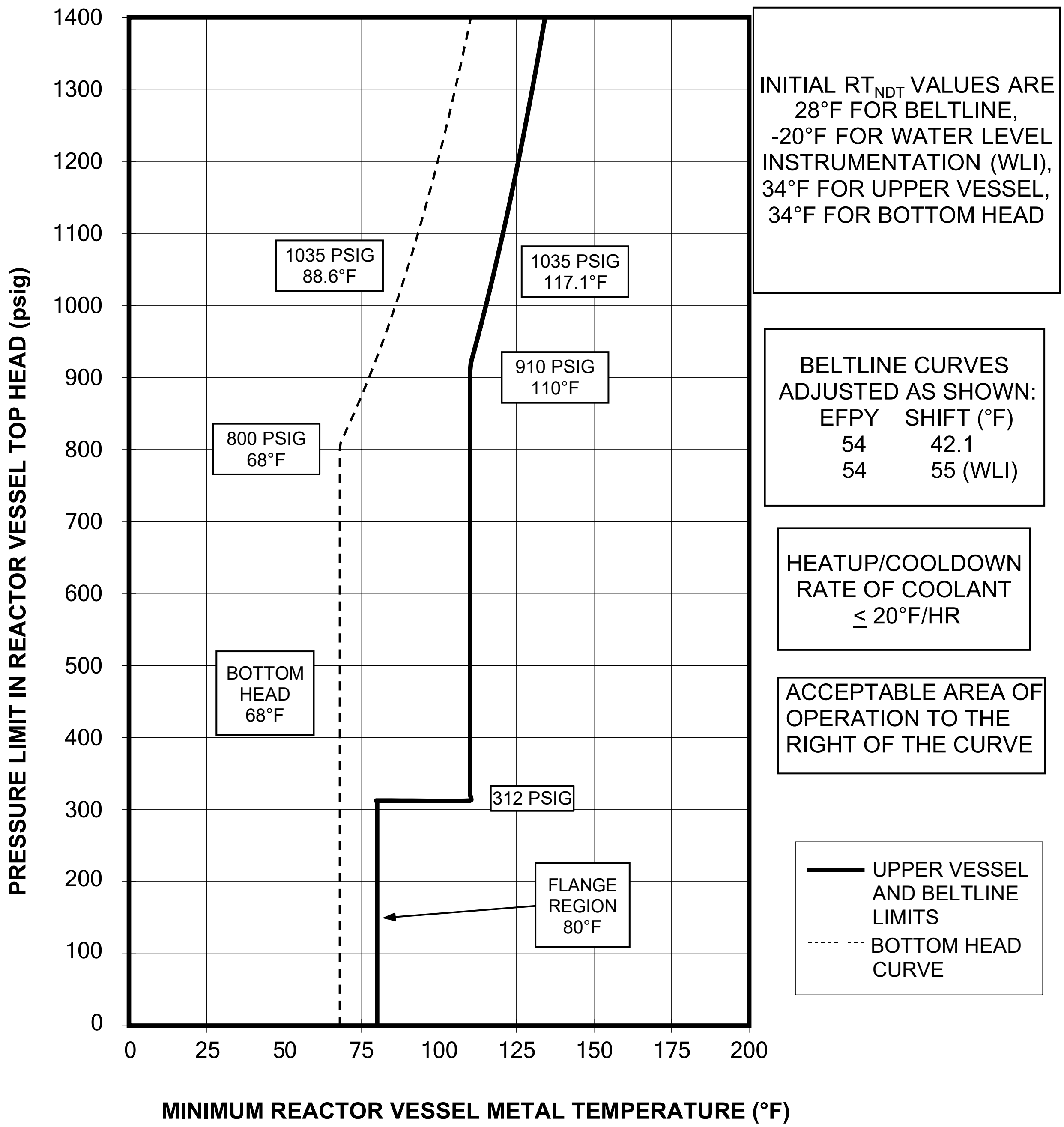

Figure 3.4.11-1 (page 1 of 1)
Inservice Leak and Hydrostatic Testing Curve

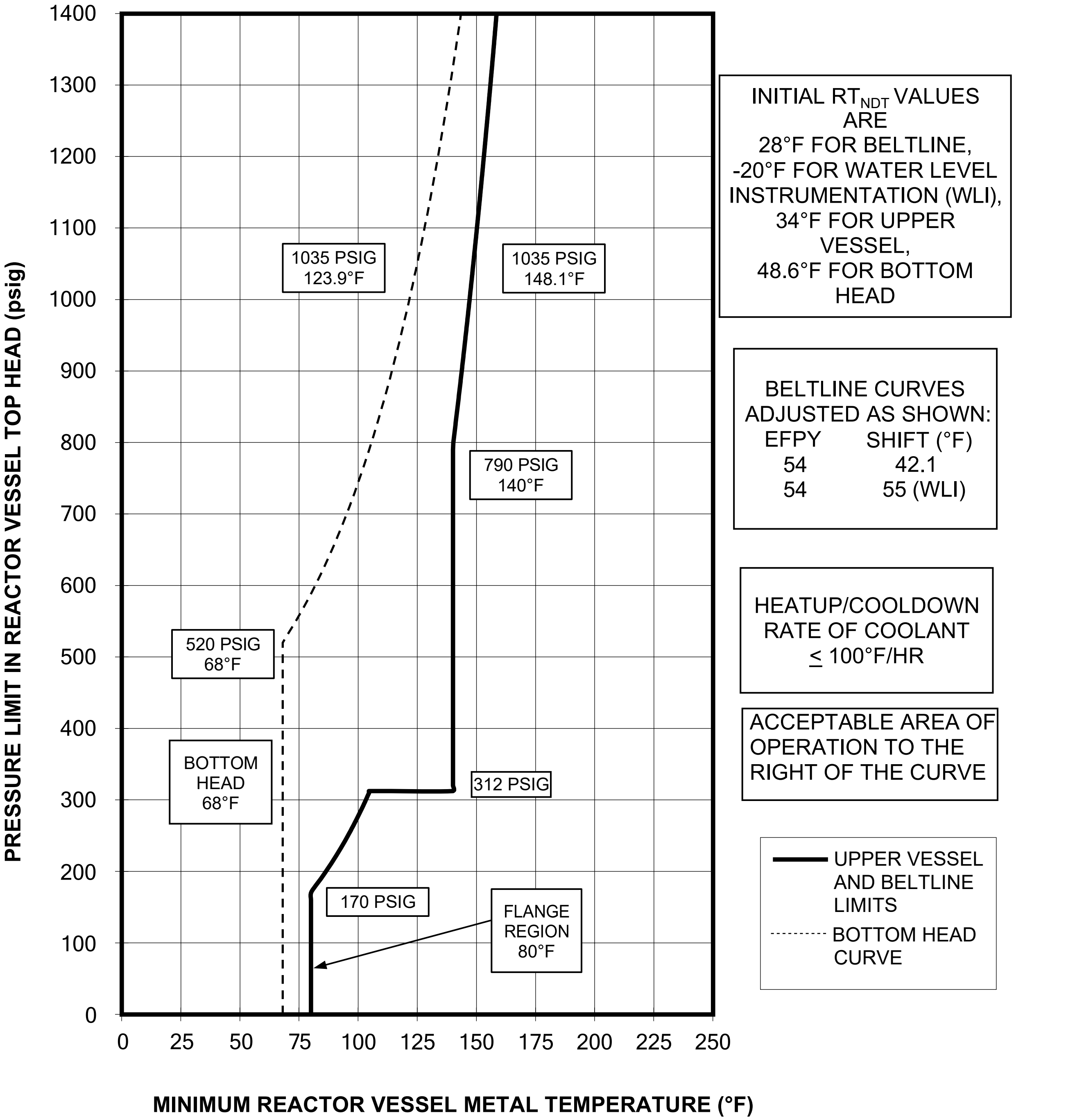

Figure 3.4.11-2 (page 1 of 1)
Non-Nuclear Heating and Cooldown Curve

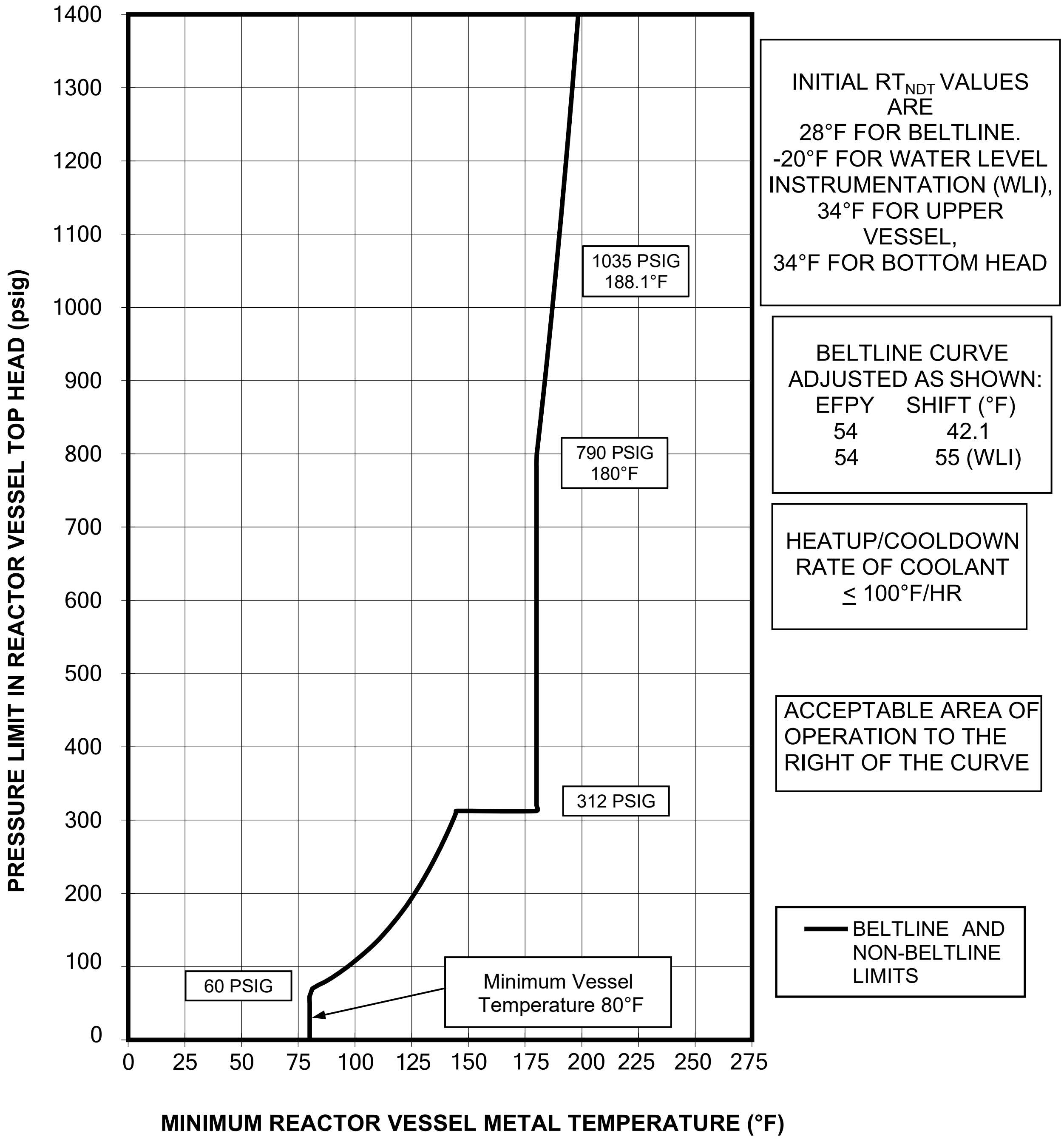

Figure 3.4.11-3 (page 1 of 1)
Nuclear Heating and Cooldown Curve

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 268 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated October 13, 2021 (Agencywide Documents and Access Management System (ADAMS) Accession Nos. ML21299A182 (public) and ML21299A183 (nonpublic)), as supplemented by letter dated July 11, 2022, (ML22192A210, (public) and ML22192A211 (nonpublic)), Energy Northwest (the licensee) submitted a license amendment request (LAR) for Columbia Generating Station (Columbia). The LAR would revise Technical Specification (TS) 3.4.11, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits," by replacing the existing RCS P/T curves in TS 3.4.11 with revised P/T curves that are projected to the end of the period of extended operation. In addition, the licensee provided the revised P/T curves to satisfy license renewal commitment number 54 in table A-1 of appendix A to NUREG-2123, "Safety Evaluation Report Related to the License Renewal of Columbia Generating Station," which the U.S. Nuclear Regulatory Commission (NRC, the Commission) published in two volumes in May 2012 (ML12139A300 and ML12139A302, respectively).

The supplemental letter dated July 11, 2022, 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 as published in the *Federal Register* on March 8, 2022 (87 FR 13017).

2.0 REGULATORY EVALUATION

2.1 Purpose of P/T Limit Curves

The purpose of the P/T limit curves in the plant TSs is to control and limit pressure, temperature, and the heatup and cooldown rates of coolant temperature in the reactor pressure vessel (RPV), such that the stresses generated from P/T during plant evolution will not cause flaws, if they exist, in the RPV shell to propagate uncontrollably and cause the RPV to fail. The RPV shell includes the beltline, upper vessel, vessel head closure assembly, attached nozzles, and bottom head. In the fracture mechanics calculation to construct P/T limit curves, the applied stresses are converted to the applied stress intensity factor, K_I, which is required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” to be less than the fracture toughness of the RPV material, K_{Ir}, so that the RPV shell material can resist propagation of the flaw(s). The higher the K_{Ir} value, the more assurance that the RPV material will resist and arrest the flaw propagation. However, the fracture toughness, K_{Ir}, of the RPV material is reduced by the neutron fluence irradiation throughout the plant operating life because of the irradiation embrittlement phenomenon. The irradiation embrittlement phenomenon is addressed via the adjusted reference temperature (ART) calculation which includes various inputs, such as the neutron fluence projection, chemistry, initial nil-ductility reference temperature (RT_{NDT}), and reactor vessel material surveillance program.

2.2 Description of the Proposed Changes

The LAR proposes to replace the P/T curves for inservice leak and hydrostatic testing, non-nuclear heating and cooldown, and nuclear heating and cooldown currently illustrated as TS figures 3.4.11-1, 3.4.11-2, and 3.4.11-3, respectively. The P/T curves proposed in the LAR are based on analyses projected to the end of the period of extended operation to satisfy license renewal commitment number 54 in table A-1 of appendix A to NUREG-2123.

2.3 Applicable Regulatory Requirements

Under 10 CFR 50.36, “Technical specifications,” paragraph (a)(1), each operating license application for a production or utilization facility must include proposed TSs and a summary statement of the bases for such specifications. Under 10 CFR 50.36(c), TSs must include the following categories related to facility operation: (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.

The regulations in 10 CR 50.36(c)(2), “Limiting conditions for operation,” states, in part, that:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

The regulation in 10 CFR 50.36(c)(3), “Surveillance requirements,” states that:

> Surveillance requirements are requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Under 10 CFR 50.60, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” all operating light-water nuclear power reactors must meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

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

Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 14, “Reactor coolant pressure boundary,” states that:

> The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Appendix A to 10 CFR Part 50, GDC 30, “Quality of reactor coolant pressure boundary,” states that:

> Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Appendix A to 10 CFR Part 50, GDC 31, “Fracture prevention of reactor coolant pressure boundary,” states that:

> The reactor coolant pressure boundary 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.

2.4 Applicable Guidance

Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” dated May 1988 (ML003740284), describes procedures for calculating the adjusted nil-ductility transition reference temperature RT_{NDT} (ART) due to neutron irradiation on RPVs.

RG 1.190, Revision 0, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001 (ML010890301), provides guidance on methods for the neutron fluence calculation. Fluence calculations are acceptable if they are performed with NRC-approved methods or with methods which are otherwise shown to conform with the guidance in RG 1.190 consistent with GDCs 14, 30, and 31.

Regulatory Issue Summary (RIS) 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” dated October 14, 2014 (ML14149A165), provides evaluation guidance for P/T curves and P/T limits reports, including the consideration of neutron fluence and structural discontinuities in the development of P/T curves.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Section 5.3.2, Revision 2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock,” dated March 2007 (ML070380185), provides an acceptable method for determining the P/T curves based on the methodology of ASME Code, Section XI, appendix G.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the following key topics in the proposed P/T curves: (1) the ART calculations, (2) the data points on the P/T curves, and (3) the proposed changes to TS 3.4.11. The NRC staff evaluated the licensee’s ART based on the neutron fluence, chemistry, initial RT_{NDT}, and use of the data from the integrated surveillance program (ISP). The NRC staff evaluated whether the proposed P/T curves appropriately address the issue of thickness discontinuity among the regions of the RPV shell. Using the ART, the NRC staff performed independent calculations to verify the data points on the licensee proposed P/T curves based on requirements of ASME Code, Section XI, appendix G and 10 CFR Part 50, Appendix G. Lastly, the NRC staff evaluated the licensee’s implementation of the proposed P/T curves in TS 3.4.11.

3.1 Licensee Proposed P/T Curves

The P/T curves in TS figure 3.4.11-1 are applicable to inservice leak and hydrostatic testing operation. The P/T curves in TS figure 3.4.11-2 are applicable to non-nuclear heating and cooldown operation. The P/T curve (one curve) in TS figure 3.4.11-3 is applicable to nuclear heating and cooldown operation. The licensee generated the proposed P/T curves by enveloping the most restrictive P/T curves from regions of the bottom head, beltline, upper vessel, and closure assembly.

The licensee established P/T curves based on the requirements of 10 CFR Part 50, Appendix G to prevent brittle fracture of the RPV. Part of the licensee’s analysis involved in developing the P/T curves is to account for irradiation embrittlement effects in the core (beltline) region. The licensee stated that it used the methods described in RG 1.190, Revision 0 and RG 1.99, Revision 2 to account for irradiation embrittlement. The licensee indicated that the key parameters, which characterize a material’s fracture toughness, are the RT_{NDT} and the upper shelf energy, which are defined in 10 CFR Part 50, Appendix G, and ASME Code, Section XI, appendix G.

The licensee stated that the proposed P/T curves are applicable to 54 effective full power years (EFPYs), representing the end of the 60-year license. The licensee’s analysis for the proposed P/T curves is shown in General Electric Hitachi (GEH) Report NEDO-33929, “Energy Northwest/Columbia Generating Station Pressure and Temperature Limits Report (PTLR) up to 54 Effective Full-Power Years,” Revision 0, which is presented in enclosure 5 (nonproprietary) and enclosure 7 (proprietary) to the LAR. The methodology of NEDO-33929 is based on the GEH Topical Report (TR) NEDC-33178P–A, Revision 1, “General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves,” dated July 29, 2009 (ML092370486). The public and non-public versions of NEDC-33178P-A are in ADAMS at ML092370487 and ML092370488, respectively. By letter dated April 27, 2009 (ML091100139), the NRC staff approved NEDC-33178P–A, Revision 1. NEDC-33178P–A has one limitation regarding the use of an approved neutron fluence calculation methodology which is addressed in section 3.2 of this safety evaluation (SE).

3.2 Neutron Fluence Calculations

Updated neutron fluence values are provided in section 3.2, "Fluence," of NEDO-33929. In section 3.0, "Technical Evaluation," of enclosure 1 to the LAR, the licensee states that the fluence projections were performed using the methods described in NRC-approved NEDC-32983P–A, Revision 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," dated January 2006 (ML072480121, public; and ML072480125, nonpublic). The NRC staff SE for NEDC-32983P–A, dated November 17, 2005 (ML053210469), indicates that the method is approved for use for boiling water reactors (BWRs), such as Columbia, based on its adherence to RG 1.190. The licensee appropriately projected fluence values supporting the proposed P/T curves to 54 EFPYs.

Based on its review, the NRC staff concludes that the information provided assures that the proposed P/T curves are based on acceptable fluence inputs because the fluence calculational methodology has been approved for use by the NRC staff for facilities like Columbia and it adheres to the guidance in RG 1.190. Therefore, the NRC staff finds that the fluence estimates described in enclosures 5 and 7 to the LAR are acceptable for use with respect to the proposed P/T curves based on a 54 EFPYs exposure period.

3.3 Chemistry

Chemistry in the context of P/T curve calculation is related to the copper and nickel contents in the RPV shell material. The copper and nickel contents are needed to calculate the ART. The licensee stated that it obtained the copper and nickel values of the RPV beltline region from Columbia vessel purchase order records, certified material test reports (CMTR), or from values previously found acceptable by the NRC staff as discussed in Section 3.1 of NEDO-33929. The licensee stated that these copper and nickel values remain unchanged from previous submittals. The licensee also considered best estimate chemistries for other relevant beltline materials from the Electric Power Research Institute (EPRI) Technical Report 3002003144, BWR Vessel and Internal Project (BWRVIP)-135, "Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations," as discussed further in this SE. The NRC staff determined that the data of BWRVIP-135 are applicable to the Columbia reactor vessel material because the surveillance material in BWRVIP-135 has the same heat number as the Columbia reactor vessel material. The NRC staff determined that the data of BWRVIP-135 are acceptable for use for the Columbia reactor vessel material because the data of BWRVIP-135 were obtained following the requirements of 10 CFR 50, Appendix H. The licensee calculated the chemistry factors for all RPV materials relevant to P/T curves based on the requirements of RG 1.99, Revision 2.

The NRC staff noted that the copper and nickel contents of various RPV components at Columbia are listed in table B-5 of appendix B to NEDO-33929. Based on these copper and nickel content values, the NRC staff performed independent calculations in accordance with RG 1.99, Revision 2 to verify the chemistry factors in table B-5. The NRC staff finds that the licensee also considered the copper and nickel contents in the material from the BWR ISP when calculating the chemistry factor per RG 1.99, Revision 2. Based on its independent calculation, the NRC staff finds that the licensee calculated the appropriate chemistry factor values in accordance with RG 1.99, Revision 2, as shown in NEDO-33929.

3.4 Initial Reference Temperature of Nil-Ductility Transition

The initial RT_{NDT} of RPV shell materials in the upper vessel, beltline, water level instrument nozzles, and bottom head are shown in TS figures 3.4.11-1, 3.4.11-2, and 3.4.11-3. The initial RT_{NDT} values for all vessel materials considered in developing the proposed P/T curves in the LAR are presented in Appendix B of NEDO-33929. The method for determining the initial RT_{NDT} for all RPV materials is defined in section 4.1 of NEDC-33178P-A. The licensee obtained the values used to determine the initial RT_{NDT} for the RPV shell material from the Columbia CMTRs. The initial RT_{NDT} for all materials remain unchanged from values previously reported for Columbia in 1997 except for the water level instrumentation nozzles (N12, N13, N14) with heat 219972 Lot 1. The licensee stated that the initial RT_{NDT} of these water level instrument nozzles were decreased to minus 20 degrees Fahrenheit (-20 °F). The updated information for the initial RT_{NDT} of N12 water level instrumentation (WLI) forging material (heat 219972 Lot 1) is shown in appendix B of NEDO-33929.

NEDC-33178P-A states that the initial RT_{NDT} is derived based on the requirements of the ASME Code, Section III, NB-2300. The NRC staff determined that regardless which edition of the ASME Code, Section III was used, the initial RT_{NDT} is derived based on the temperature data from the Charpy V-Notch specimens and drop weight nil-ductility transition specimens. The NRC staff noted that 10 CFR Part 50, Appendix G specifies additional requirements on the derivation of initial RT_{NDT} to which the licensee needs to satisfy.

Tables, B-1, B-2, B-3, and B-4 of Appendix B to NEDO-33929 provide initial RT_{NDT} of various Columbia RPV components. The NRC staff reviewed material data and performed independent calculations to verify the initial RT_{NDT} in these tables. The NRC staff verified that the initial RT_{NDT} used for the upper vessel region, beltline region, nozzles, and bottom head in the P/T curves are calculated in accordance with the ASME Code, Section III and 10 CFR Part 50, Appendix G. Therefore, the NRC staff finds that the initial RT_{NDT} used in the proposed P/T curves are acceptable.

3.5 Reactor Vessel Material Surveillance Program

The industry-wide RPV material surveillance program consists of surveillance capsules that are installed inside the RPVs of domestic BWR units. The surveillance capsule contains flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the BWR RPV materials within the core beltline region. The Columbia Updated Final Safety Analysis Report (ML19343C030) Section 5.3.1.6, “Material Surveillance,” states that the BWRVIP ISP will determine the removal schedule for the remaining Columbia surveillance capsules. The licensee stated that the Columbia material surveillance program is administered in accordance with the BWRVIP ISP which is administered by EPRI and the BWR Owners’ Group. The ISP combines the domestic BWR surveillance programs into a single integrated program. This program uses similar heats of materials in the surveillance programs of various BWRs nuclear plants to represent the limiting materials in other BWR RPVs.

Appendix A of NEDO-33929 states that in accordance with 10 CFR Part 50, Appendix H, the licensee removed the first surveillance capsule from the Columbia RPV after 7.2 EFPYs during refueling outage R11 in 1996. The licensee tested and reconstituted the specimens in the capsule and returned the capsule to the RPV in refueling outage R12 (1997). Subsequently, the licensee found that the same specimen holder failed in refueling outage R23 (2017) and removed it from the Columbia RPV. Section 3.5 of NEDO-33929 states that two of the surveillance capsules installed at plant startup remain in the RPV. The licensee stated that Columbia is not a host plant for the ISP. Therefore, the three surveillance capsules are designated as deferred per NRC staff-approved EPRI Technical Report 1025144, BWRVIP-86, Revision 1-A, “Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,” May 2013 (ML131760082). The licensee further stated that because of the deferral, BWRVIP-135 provides the representative surveillance data considered in determining the chemistry and any fitted or adjusted chemistry factors for the beltline materials for Columbia.

The licensee stated that the ISP representative welds, heat numbers 5P6756 and 5P6214B, were considered in its ART calculations in developing the P/T curves. The ISP representative weld heat number 5P6756 is the identical heat number as the Columbia lower shell and lower intermediate circumferential weld heat. The ISP representative weld heat number 5P6214B is the identical heat as the heat number for the Columbia beltline N6 low pressure coolant injection (LPCI) nozzle weld. The licensee stated that the maximum scatter in the fitted data falls within the 1-sigma value of 28 °F from RG 1.99, Revision 2. The licensee explained that as the ISP weld material is identical to both Columbia vessel welds, the data from both ISP weld materials were used to calculate an adjusted chemistry factor to reduce the margin term as defined in RG 1.99, Regulatory Position C.2.1. The licensee used the adjusted chemistry factors to derive the ART values for these two Columbia welds using the ISP weld data.

The NRC staff recognized that the RPV material surveillance program at Columbia is part of the industry wide BWR ISP that is administered by EPRI and the BWR Owners’ Group. The NRC staff finds that, as discussed in section 3.5 of NEDO-33929, the licensee has used various data from the materials in the BWR ISP to derive the chemistry factors in accordance with RG 1.99, Revision 2. As such, the NRC staff confirmed that the RPV material surveillance program at Columbia is acceptable. The NRC staff finds that the licensee used the chemistry in the BWR ISP material to calculate the chemistry factor in accordance with RG 1.99, Revision 2. Based on its evaluation, the NRC staff determined that Columbia’s surveillance program is implemented in accordance with 10 CFR Part 50, Appendix H because the licensee has followed Section III of Appendix H, including paragraph III.C.1.

3.6 Adjusted Reference Temperature

The ART values of the Columbia RPV materials for 54 EFPYs are presented in table B-5 of appendix B to NEDO-33929. The licensee stated that the ART values in table B-5 considered the latest BWRVIP ISP published surveillance data available that is representative of the applicable materials in the Columbia RPV. The licensee updated the chemistry factor value for weld with ISP data and used the chemistry factor value for the determination of the ART. The licensee explained that as the N12 WLI weld material is Inconel, for which fracture toughness evaluations are not required, only the ART for N12 WLI forging material is evaluated. The licensee evaluated the N6 LPCI nozzle weld and N12 WLI forging material for fracture toughness.

The NRC staff verified that the ART values for various Columbia RPV materials in table B-5 of appendix B to NEDO-33929 are calculated in accordance with RG 1.99, Revision 2 and are therefore finds the ART values acceptable. The NRC staff finds that the licensee has used the ART value of the limiting material to construct the proposed P/T curves such that irradiation embrittlement of the RPV material is appropriately accounted for in the proposed P/T curves for 54 EFPYs.

3.7 Thickness Discontinuities

The difference in thickness of various regions of the RPV shell may cause stresses to increase at the junction of two components. The stresses due to thickness discontinuity should be considered in the fracture mechanics calculations to develop the P/T curves.

The licensee stated that the Columbia RPV has four thickness discontinuities that need to be addressed in the P/T curve evaluation: (1) lower shell to bottom head torus, (2) lower shell to lower intermediate shell, (3) upper intermediate shell to upper shell, and (4) bottom head radial plate to bottom head dollar plate. The licensee evaluated the vessel wall thickness discontinuities to ensure that the proposed P/T curves follow the guidance of RIS 2014-11 due to the beltline thickness discontinuities. The licensee also considered the thickness discontinuity between the top head dollar plate and torus. The licensee calculated the P/T curves for the top head considering this thickness continuity to ensure that the RPV is adequately protected, or "bounded."

The NRC staff finds that, as shown in NEDO-33929, the licensee analyzed the thickness of various regions of the RPV in its proposed P/T curve calculations following the guidance of RIS 2014-11 and requirements of 10 CFR 50, Appendix G. Therefore, the NRC staff finds that the proposed P/T curves have appropriately addressed the issue of thickness discontinuity among the regions of the RPV shell because it meets SRP 5.3.2 and the requirements of 10 CFR 50, Appendix G.

3.8 Evaluation of Proposed P/T Curves

ASME Code, Section XI, appendix G, requires a fracture mechanics calculation (or P/T curve calculation) to develop the P/T curves. The calculation postulates two flaws in the RPV shell with each flaw having a depth of 1/4T (T = the wall thickness of the RPV shell) and varying length. One flaw is postulated to initiate from the interior surface of the RPV growing toward exterior surface (i.e., the 1/4T location flaw). Another flaw is postulated to initiate from the exterior surface of the RPV growing toward interior surface (i.e., the 3/4T location flaw). The applied pressure and thermal stresses are calculated for the 1/4T and 3/4T location flaws. The tensile stress due to thermal gradient is in the inner wall during cooldown and the outer wall during heatup. However, as a conservative simplification, the licensee assumed the thermal gradient stress at the 1/4T location to be tensile for both heatup and cooldown. The licensee stated that this approach results in the maximum tensile stress at the 1/4T location and is conservative because irradiation effects cause the allowable toughness, K_{Ir}, at 1/4T to be less than at 3/4T for a given metal temperature. The NRC staff determined that the lower the K_{Ir} value, the more conservative the P/T curves. The NRC staff recognizes that it is a conservative approach when the thermal gradient stress at the 1/4T is assumed to be tensile for both heatup and cooldown. Therefore, the NRC staff finds that the licensee's assumption regarding the thermal stress calculation at the 1/4T location is acceptable.

The NRC staff determined that the licensee included the hydrostatic pressure of the coolant in the RPV based on the height of the vessel and the elevation of the RPV region in question (e.g., the beltline region and bottom head) in its proposed P/T curve calculation. The NRC staff determined that the licensee included the internal pressure and static pressure head to derive the applied K_I for the development of the proposed P/T curves in various RPV regions as discussed in section 4.3.2 of NEDC-33178P–A, Revision 1.

The NRC staff determined that specific calculations are needed to derive stress distributions of the complex geometry such as nozzles and the bottom head, which contains many control rod drive nozzles that are attached to the RPV. The NRC staff verified that NEDC-33178P–A, Revision 1, uses finite element models as part of specific calculations to derive stress distributions of various nozzles such as WLI, feedwater and bottom head nozzles. During the review of NEDC-33178P-A, the NRC staff reviewed plant-specific P/T limit curves that the NRC staff approved previously that used the methodology of the TR. As part of the TR review, the staff verified that the previously NRC staff-approved plant-specific P/T limit curves have used the TR methodology appropriately. As part of those previous plant-specific NRC reviews, the staff verified the plant-specific P/T limit curves for the beltline region by performing independent calculations to support its conclusion that the TR satisfied 10 CFR Part 50, Appendix G and ASME Code, Section XI, Appendix G. Additionally, the staff verified whether the analysis results using the methodologies in Section 4.3 and Appendices F, G and H of the TR were applicable to and bounded the plant-specific geometry and loading conditions. For example, for the bottom head P/T limit curves, the staff verified whether the R/t ratio (where R = radius of the bottom head, and t = bottom head wall thickness) of the plant-specific bottom head is within the allowable R/t ratio that was used in the analysis of Appendix H of the TR. If the R/t ratio of the plant-specific bottom head satisfied the allowable ratio, the analytical results of the Appendix H of the TR were acceptable for use in the development of the plant-specific P/T limit curves.

Based on the evaluation performed in those previously approved plant-specific reviews, the staff determined that the previously approved plant-specific P-T limit curves satisfied 10 CFR Part 50, Appendix G and the ASME Code, Section XI, Appendix G. The staff concluded that the previously NRC staff-approved plant-specific P/T limit curves demonstrated that the methodology of NEDC-33178P-A is acceptable for generic use. The methodology of the TR has defined criteria and characteristics to address the operating plants in the BWR fleet. The NRC staff determined that Columbia has used the appropriate methodology criteria in the TR to develop its plant-specific P/T limit curves. The licensee has shown and the staff has confirmed that the appropriate methodology criteria were used in the evaluation of the WLI, feedwater and bottom head.

The NRC staff performed independent calculations to verify the proposed P/T curves in TS figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 based on information provided in NEDC-33178P–A, Revision 1; NEDO-33929, Revision 0; the ASME Code, Section XI, appendix G; SRP 5.3.2; and 10 CFR Part 50, Appendix G.

The NRC staff evaluated the beltline and non-beltline regions of the proposed P/T curves as discussed below.

3.8.1 P/T Curves in the Beltline Region

For the beltline region, the NRC staff verified that the licensee used the ART of the limiting material for 54 EFPYS to develop the proposed P/T curves. The NRC staff further verified that the licensee also considered the ART from other components in the beltline region such as the N6 LPCI nozzle and the N12 WLI nozzle in the development of the proposed P/T curves. The nozzles that are attached to the beltline region require use of the finite element analysis to determine the stress distribution because of the unique nozzle geometry. The NRC staff determined that stresses developed from the generic finite element analysis in NEDC-33178P-A were used to generate a specific curve applicable for the WLI nozzle to ensure that it is bounded in the proposed P/T curves. The NRC staff verified that the Columbia N12 WLI nozzle is similar to that shown in figure 1 in appendix J of NEDC-33178P-A.

Therefore, the NRC staff finds that the proposed P/T curves have included the limiting material and stress distributions of beltline regions in the development of the proposed P/T curves as required by 10 CFR 50, Appendix G.

3.8.2 P/T Curves in the Non-Beltline Region

Appendix G to 10 CFR Part 50 requires that the P/T curves be developed for the ferritic materials not in the RPV beltline region. In addition, NRC RIS 2014-11 clarifies that P/T curves for ferritic RPV components that are not beltline shell materials may define P/T curves that are more limiting than those calculated for the RPV beltline shell materials. For the non-beltline regions, the licensee considered the ART of various components in the bottom head region and upper vessel region in developing the proposed P/T curves. For example, feedwater and other system nozzles and closure flange are part of the upper vessel region. The control rod drive and drain line penetrations are part of the bottom head region. The licensee stated that the feedwater nozzle is the limiting material for the upper vessel (non-beltline) region because of its stress distributions. Consistent with RIS 2014-11 and 10 CFR 50, Appendix G, the NRC staff determined that the licensee's finite element analyses, as shown in NEDC-33178P–A, Revision 1, used thermal and pressure stress distributions in the feedwater nozzle, from which pressure stress intensity factor (K_{Im}) and thermal stress intensity factor (K_{It}) were derived to construct the P/T curves

With regard to the bottom head region, the NRC staff determined that the generic analysis of the BWR vessel model in NEDC-33178P–A, Revision 1 is applicable to the Columbia BWR vessel based on parameter $R/\sqrt{t}$, where "R" is the bottom head radius, and "t" is the bottom head wall thickness. The NRC staff notes that proposed TS figures 3.4.11-1 and 3.4.11-2 specifically include bottom head P/T limit curves to protect the bottom head which contains control rod drive penetration nozzles. In its July 11, 2022, letter, the licensee stated that the bottom head temperature is monitored based on the bottom head drain line with the drain line flow established.

The NRC staff notes that the initial RT_{NDT} for the bottom head in proposed TS figure 3.4.11-2 is higher than that of proposed TS figures 3.4.11-1 and 3.4.11-3. In its July 11, 2022, letter, the licensee explained that a higher temperature was used for the initial RT_{NDT} of the bottom head in proposed TS figure 3.4.11-2 to provide additional operational flexibility because of emergent events. The NRC staff determined that the initial RT_{NDT} temperature is acceptable for the initial RT_{NDT} for the bottom head in TS figure 3.4.11-2 because the higher temperature used for the RT_{NDT} provides operational flexibility.

Therefore, the NRC staff finds that the licensee analyzed relevant non-beltline regions as part of the proposed P/T curve development in accordance with RIS 2014-11 and 10 CFR 50, Appendix G.

3.8.3 Evaluation of Specific P/T Curves

The NRC staff evaluated each proposed P/T curve in accordance with 10 CFR Part 50, Appendix G, and the ASME Code, Section XI, appendix G, as discussed below.

The licensee used the methodology in TR EDC-33178P–A, Revision 1 to calculate the P/T curves. The staff evaluated the generic methodology in Section 4.3 of NEDC-33178P–A, Revision 1 that requires the minimum temperature of the P/T curves follow the minimum temperature requirements in Table 1 of 10 CFR Part 50, Appendix G. The staff has found the methodology in Section 4.3 of the TR for calculating the minimum temperature acceptable because the generic methodology of the TR satisfies the prescriptive minimum temperature requirements of Table 1 in 10 CFR Part 50, Appendix G. In addition, the staff noted that the previously NRC staff-approved plant-specific P/T limit curves do incorporate the required minimum temperatures based on the generic methodology of Section 4.3 of the TR in accordance with Table 1 of 10 CFR Part 50, Appendix G. Based on its evaluation, the staff finds that the generic methodology of the TR addresses the minimum temperature requirement of 10 CFR Part 50, Appendix G satisfactorily and therefore is acceptable for use. Therefore, the NRC staff finds the use of NEDC-33178P–A, Revision 1 acceptable to calculate the P/T curves for Columbia.

3.8.3.1 Inservice Leak and Hydrostatic Testing Curve (Curve A)

The NRC staff determined that proposed Curve A in TS figure 3.4.11-1 for the inservice leak and hydrostatic testing is based on a temperature rate of less than or equal to (≤) 20 °F/hour. As such, the thermal stress is considered negligible. Therefore, the licensee set the K_{It} to zero in the fracture mechanics calculation. The NRC staff finds acceptable that the K_{It} is zero because temperature gradient in the RPV shell wall is too small to generate any significant thermal stresses during the hydrostatic and leakage tests. The NRC staff determined that licensee appropriately set the safety factor for the K_{Im} to 1.5 in accordance with the requirements of the ASME Code, Section XI, Appendix G.

Appendix G to 10 CFR Part 50, Table 1, item 1.a, requires that when the operating pressure is ≤ 20 percent of the preservice system hydrostatic test pressure, the minimum temperature on the P/T curve must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload. The NRC staff confirmed that the preservice system hydrostatic test pressure used in the licensee’s proposed P/T curve calculation is consistent with P/T curves in other BWR plants and consistent with the requirements of 10 CFR 50, Appendix G and guidance of SRP 5.3.2 and therefore is acceptable. The NRC staff verified that the proposed Curve A in TS figure 3.4.11-1 satisfies 10 CFR Part 50, Appendix G, Table 1, item 1.a.

Appendix G to 10 CFR Part 50, Table 1, item 1.b, requires that when the pressure is greater than (>) 20 percent of the preservice system hydrostatic test pressure, the minimum temperature on the P/T curve must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload plus 90 °F. The item 1.b requirement creates a bend (a knee) in the non-beltline P/T curve at 312.6 pounds per square inch gauge (psig) because the closure flange is part of the non-beltline region. The NRC staff verified that the proposed Curve A of TS figure 3.4.11-1 satisfies 10 CFR Part 50, Appendix G, Table 1, item 1.b. Therefore, the NRC staff finds proposed Curve A in TS figure 3.4.11-1 acceptable because it satisfies the requirements of items 1.a and 1.b of 10 CFR Part 50, Appendix G, Table 1 and the ASME Code, Section XI, appendix G.

3.8.3.2 Non-Nuclear Heating and Cooldown Curve (Curve B)

The NRC staff determined that Curve B in TS figure 3.4.11-2 is applicable during times when the coolant heatup or cooldown rate is greater than 20 °F/hour during a pressure test and when the core is not critical. Additionally, Curve B must be followed when performing low-power physics testing. The heatup and cooldown rate is limited to ≤ 100 °F/hour when using Curve B.

The NRC staff verified that in accordance with the ASME Code, Section XI, appendix G to construct proposed Curve B, the licensee used the safety factors of 2.0 and 1.0 for the K_{Im} and K_{It}, respectively.

Appendix G to 10 CFR Part 50, Table 1, item 2.a requires that when the operating pressure is ≤ 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload. The NRC staff verified that the proposed Curve B in TS figure 3.4.11-2 satisfies item 2.a.

Appendix G to 10 CFR Part 50, Table 1, item 2.b, requires that when the operating pressure is > 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload plus 120 °F. The item 2.b requirement creates a bend for the non-beltline P/T curve at 312.6 psig because the closure flange is part of the non-beltline region. The NRC staff verified that the proposed Curve B in TS figure 3.4.11-2 satisfies item 2.b.

Therefore, the NRC staff finds proposed Curve B in TS figure 3.4.11-2 acceptable because it satisfies the requirements of items 2.a and 2.b in 10 CFR Part 50, Appendix G, Table 1 and the ASME Code, Section XI, appendix G.

3.8.3.3 Nuclear Heating and Cooldown Curve (Curve C)

The NRC staff determined that the proposed Curve C in TS figure 3.4.11-3 is used when the core is critical with the exception during low-power physics testing activities. The heatup and cooldown rate is limited to ≤ 100 °F/hour when using proposed Curve C. The NRC staff verified that in accordance with the ASME Code, Section XI, appendix G to construct proposed Curve C, the licensee used the safety factor of 2.0 and 1.0 for the K_{Im} and K_{It}, respectively.

Appendix G to 10 CFR Part 50, Table 1, items 2.c, 2.d, and 2.e require that the P/T curve for the core critical condition (i.e., Curve C) be 40 °F greater than the P/T curves for the core not critical (i.e., Curve B) under all operating pressure conditions. The NRC staff verified that proposed Curve C is 40 °F greater than proposed Curve B and that the proposed Curve C in TS figure 3.4.11-3 satisfies the 40 °F requirement of items 2.c, 2.d, and 2.e.

Regarding the minimum temperature requirement, Appendix G to 10 CFR Part 50, Table 1, item 2.c requires that when the operating pressure is ≤ 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be larger of the minimum permissible temperature for the inservice system hydrostatic pressure test or the closure flange RT_{NDT} plus 40 °F. The NRC staff verified that the proposed Curve C in TS figure 3.4.11-3 satisfies item 2.c.

Appendix G to 10 CFR Part 50, Table 1, item 2.d requires that when the reactor internal pressure is > 20 percent of the preservice system hydrostatic test pressure, the minimum temperature must be larger of the minimum permissible temperature for the inservice system hydrostatic pressure test or closure flange RT_{NDT} plus 160 °F. The NRC staff finds that the notch exists in proposed Curve C at 312.6 psig and item 2.d requirement is satisfied.

Appendix G to 10 CFR Part 50, Table 1, item 2.e requires that for BWRs, 60 °F be added to the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload when the operating pressure is less than (<) 20 percent of preservice system hydrostatic test pressure. The NRC staff verified that the proposed Curve C in TS figure 3.4.11-3 satisfies item 2.e.

The NRC staff finds proposed Curve C in TS figure 3.4.11-3 acceptable because it satisfies the requirements of sections IV.2.c, IV.2.d, and IV.2.e of 10 CFR Part 50, Appendix G, Table 1 and the ASME Code, Section XI, Appendix G.

3.9 Beltline Weld Flaw Indications

Section 3.6 of NEDC-33929 states that Columbia has two indications near the RPV shell welds BG and BM. The NRC staff noted that the proposed P/T curves are developed based on the assumption that flaws exist in the RPV shell. The NRC staff questioned whether these two indications/flaws affect the postulated flaw in the proposed P/T curves calculations. In the July 11, 2022, letter, the licensee stated that the two indications are subsurface flaws. The licensee stated that the flaw near the BM weld is not recordable. The licensee evaluated the impact of the flaw near the BG weld on the proposed P/T curves and concluded that the flaw does not affect the proposed P/T curves. The NRC staff compared the size of the actual flaw with the flaw size assumed in the proposed P/T curve calculations and determined that the actual flaw in the RPV shell does not affect the proposed P/T curves.

3.10 Evaluation of Changes to TSs

The LCO statement for TS 3.4.11.1 states: "RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation loop temperature requirements shall be maintained within limits." SR 3.4.11.1 requires verification that RCS pressure and RCS temperature are within the applicable limits specified in figures 3.4.11-1, 3.4.11-2, and 3.4.11-3. SR 3.4.11.2 requires verification that RCS pressure and RCS temperature are within the criticality limits specified in figure 3.4.11-3. These TS requirements ensure that operation is within the limits of the curves in figures 3.4.11-1, 3.4.11-2, and 3.4.11-3 and will ensure that the stresses generated from pressure and temperature during plant evolutions will not cause flaws, if they exist, in the RPV shell to propagate uncontrollably and cause a catastrophic failure

The NRC staff finds the proposed changes acceptable because the TSs, as amended by the proposed changes, will continue to meet the requirements of 10 CFR 50.36(c)(2) and (c)(3) because the LCOs will continue to be the lowest functional capability or performance levels of equipment required for safe operation of the facility and the SRs will continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that LCO will be met.

3.11 Technical Evaluation Summary

The NRC staff has determined, for the reasons discussed above, that the proposed P/T curves were developed based on the limiting ART, which was appropriately derived based on RG 1.99, Revision 2 and RG 1.190, Revision 0. The proposed P/T curves considered stresses in the beltline and non-beltline regions. The NRC staff has determined that the proposed P/T curves in Columbia TS section 3.4.11 are appropriately constructed satisfying the methodology of the ASME Code, Section XI, appendix G. The NRC staff also finds that the minimum temperature limits in the proposed P/T curves satisfy the requirements of 10 CFR Part 50, Appendix G. Accordingly, the NRC staff finds that the P/T curves continue to meet the requirements of 10 CFR 50.60, and GDCs 14, 30, and 31.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Washington State official was notified of the proposed issuance of the amendment on September 16, 2022. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 published in the *Federal Register* on March 8, 2022 (87 FR 13017), and there has been no public comment on such finding. 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: Matt Hamm
Ashley Smith
John Tsao

Date: November 23, 2022

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT NO. 268 TO REVISE TECHNICAL SPECIFICATION 3.4.11 “RCS PRESSURE AND TEMPERATURE (P/T) LIMITS” (EPID L-2021-LLA-0191) DATED NOVEMBER 23, 2022

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OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DNRL/NVIB/BC
NAME	MChawla (DGalvin for)	PBlechman	ABuford
DATE	9/22/2022	9/22/2022	9/8/2022
OFFICE	NRR/DSS/SCPB/BC	NRR/DSS/SFNB/BC	NRR/DSS/STSB/BC
NAME	BWittick	SKrepel	VCusamano
DATE	9/21/2022	9/20/2022	9/23/2022
OFFICE	OGC - NLO	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM
NAME	JEzell	JDixon-Herrity	MChawla (DGalvin for)
DATE	11/22/2022	11/23/2022	11/23/2022

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