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

April 14, 2021

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

SUBJECT: COLUMBIA GENERATING STATION – APPROVAL FOR RELIEF REQUEST 4ISI-09, REGARDING ALTERNATE EXAMINATION OF FEEDWATER NOZZLES (EPID L-2020-LLR-0068)

Dear Mr. Sawatzke:

By letter dated April 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20114E235), as supplemented by letter dated October 22, 2020 (ADAMS Accession No. ML20296A684), Energy Northwest (the licensee) requested relief from the inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1 at Columbia Generating Station (Columbia).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(1), the licensee submitted Relief Request 4ISI-09 for the alternate examination of reactor vessel feedwater nozzle inner radii and associated nozzle-to-shell welds (feedwater nozzle assemblies) on the basis that the proposed alternative would provide an acceptable level of quality and safety. Specifically, the proposed alternative would reduce the number of feedwater nozzles examined from 100 percent of the feedwater nozzles to 25 percent of the feedwater nozzles.

Based on the licensee's information submitted, the U.S. Nuclear Regulatory Commission (NRC) staff determined that the proposed alternative provides reasonable assurance of structural integrity of the subject feedwater nozzle inner radii and associated nozzle-to-shell welds and, therefore, provides an acceptable level of quality and safety. As set forth in the enclosed safety evaluation, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC authorizes the use of Relief Request 4ISI-09 at Columbia for the remainder of plant life including the period of extended operation ending on December 12, 2043.

All other requirements in the ASME Code, Section XI, for which relief was not specifically requested and approved in this proposed alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions regarding this matter, please contact the Project Manager, Mahesh Chawla at (301) 415-8371 or via e-mail at Mahesh.Chawla@nrc.gov.

Sincerely,

Jennifer L. Dixon-Herrity

Digitally signed by Jennifer L. Dixon-Herrity
Date: 2021.04.14 11:54:52 -04'00'

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

Docket No. 50-397

Enclosure:
Safety Evaluation

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 4ISI-09

ALTERNATE EXAMINATION OF REACTOR VESSEL FEEDWATER NOZZLES

AND NOZZLE-TO-SHELL WELDS

COLUMBIA GENERATING STATION

ENERGY NORTHWEST

DOCKET NO. 50-397

1.0 Introduction

By letter dated April 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20114E235), as supplemented by letter dated October 22, 2020 (ADAMS Accession No. ML20296A684), Energy Northwest (the licensee) requested relief from the inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1 at Columbia Generating Station (Columbia).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1)), "Acceptable level of quality and safety," the licensee submitted Relief Request 4ISI-09 for the alternate examination of reactor vessel feedwater nozzle inner radii and associated nozzle-to-shell welds (feedwater nozzle assemblies) on the basis that the proposed alternative would provide an acceptable level of quality and safety. Specifically, the proposed alternative would reduce the number of feedwater nozzles examined from 100 percent of the feedwater nozzles to 25 percent of the feedwater nozzles.

2.0 Regulatory Evaluation

The regulations in 10 CFR 50.55a(g), "Preservice and inservice inspection requirements," require that the inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda. Section 50.55a(g)(4) of 10 CFR states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in ASME Code, Section XI.

ASME Code, Section XI, requires that all reactor pressure vessel (RPV) nozzles be inspected during each 10-year ISI interval. The volumes in each nozzle required to be inspected are 100 percent of the nozzle inner radius section volume and 100 percent of the associated nozzle-to-vessel shell weld volume, as shown in the applicable figure in Figures IWB-2500-7(a) through (d), “Nozzle in Shell or Head,” of the ASME Code, Section XI.

By letter dated November 25, 2002, the Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) submitted for U.S. Nuclear Regulatory Commission (NRC) review and approval Technical Report (TR) BWRVIP-108, “Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii,” to reduce inspection requirements of BWR reactor vessel nozzle-to-shell welds and nozzle blend radii. By letter dated December 19, 2007 (ADAMS Accession No. ML073600374), the NRC approved the use of TR BWRVIP-108 with conditions in a safety evaluation (SE).

By letter dated April 26, 2011 (ADAMS Accession No. ML11119A041), the BWRVIP submitted for NRC review and approval, BWRVIP-241, “Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii.” BWRVIP-241 provides analyses for BWR RPV recirculation inlet and outlet nozzle-to-shell welds and nozzle inner radii to address the conditions specified in the NRC’s SE for BWRVIP-108. By letter dated April 19, 2013 (ADAMS Accession No. ML13071A245), the NRC approved the use of BWRVIP-241 with conditions in an SE.

By letter dated October 10, 2012 (ADAMS Accession No. ML12290A017), the BWRVIP, as part of its response to NRC’s RAI, submitted for NRC review and approval a supplemental document to BWRVIP-108 and BWRVIP-241, Appendix A, “BWR Nozzle Radii and Nozzle-to-Vessel Welds Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21).” In the October 10, 2012, letter, the BWRVIP stated that the purpose of Appendix A is to provide the applicants to use and reference BWRVIP-108 and BWRVIP-241 in a plant-specific integrated plant assessment and time-limited aging analysis evaluation in their license renewal application to extend plant operation to 60 years.

By letter dated April 26, 2017 (ADAMS Accession No. ML17114A096), the NRC issued a Final SE approving Appendix A to BWRVIP-108 and BWRVIP-241, thus, extending the applicability of these reports to the period of extended operation to 60 years. The NRC’s Final SE requires BWR licensees to submit a relief request for prior NRC review and approval if an alternate examination from the ASME Code, Section XI requirements will be used for reactor vessel nozzles and attachment welds.

The licensee stated in Relief Request 4ISI-09 that BWRVIP-108 and BWRVIP-241 provide the technical basis to support ASME Code Case N-702, “Alternative Requirements for Boiling Water Reactor Nozzle Inner Radius and Nozzle-to-Shell Welds Section XI, Division 1,” which provides an alternative that would reduce the inspection of the number of RPV nozzles. The NRC staff notes that Relief Request 4ISI-09 is not based on ASME Code Case N-702 but is based on plant-specific analyses using the relevant methodology in BWRVIP-108, BWRVIP-241, and Appendix A to these two reports.

The NRC staff notes that BWRVIP reports that have been approved by the NRC are designated with an “A” attached to the report number (e.g., BWRVIP-108-A and BWRVIP-241-A).

Section 50.55a(z), “Alternative to codes and standards requirements,” of 10 CFR states that:

> Alternatives to the requirements of paragraphs (b) through (h) of [10 CFR 50.55a] or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and

authorized prior to implementation. The applicant or licensee must demonstrate that:

(1) *Acceptable level of quality and safety*. The proposed alternative would provide an acceptable level of quality and safety; or

(2) *Hardship without a compensating increase in quality and safety*. Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of an alternative and the NRC to grant relief and the use of the proposed alternative.

3.0 Technical Evaluation

3.1 Licensee's Proposed Alternative

3.1.1 ASME Code Component(s) Affected

The affected components are ASME Code Class 1 RPV feedwater (RFW) nozzles as identified in Table 1 below. The feedwater nozzles and associated nozzle-to-shell welds are classified as Examination Category B-D, Item Nos. B3.90, “Nozzle-to-Vessel Welds,” and B3.100, “Nozzle Inside Radius Section,” in Table IWB-2500-1, “Examination Category B-D, Full Penetration Welded Nozzles in Vessels.” of the ASME Code, Section XI.

Table 1

Identification Number	Description	Code Category	Item Number
N4-30	RFW Nozzle-to-Shell Weld @ 30 Degrees	B-D	B3.90
N4-30-IR	RFW Nozzle Inner Radius @ 30 Degrees	B-D	B3.100
N4-90	RFW Nozzle-to-Shell Weld @ 90 Degrees	B-D	B3.90
N4-90-IR	RFW Nozzle-to-Shell Weld @ 90 Degrees	B-D	B3.100
N4-150	RFW Nozzle-to-Shell Weld @ 150 Degrees	B-D	B3.90
N4-150-IR	RFW Nozzle Inner Radius @ 150 Degrees	B-D	B3.100
N4-210	RFW Nozzle-to-Shell Weld @ 210 Degrees	B-D	B3.90
N4-210-IR	RFW Nozzle Inner Radius @ 210 Degrees	B-D	B3.100
N4-270	RFW Nozzle-to-Shell Weld @ 270 Degrees	B-D	B3.90
N4-270-IR	RFW Nozzle Inner Radius @ 270 Degrees	B-D	B3.100
N4-330	RFW Nozzle-to-Shell Weld @ 330 Degrees	B-D	B3.90
N4-330-IR	RFW Nozzle Inner Radius @ 330 Degrees	B-D	B3.100

3.1.2 Applicable Code Edition And Addenda

The licensee stated in its letter dated April 22, 2020, that the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI is the Code of Record for the fourth 10-year ISI interval for Columbia. For ultrasonic examinations, the licensee has implemented ASME Code, Section XI, Appendix VIII, “Performance Demonstration for Ultrasonic Examination Systems,” as required and as modified by 10 CFR 50.55a.

3.1.3 Applicable Code Requirement

The applicable Code requirement for inspection is contained in ASME Code Section XI, Division 1, Subsection IWB, Table IWB-2500-1. The examination requirements for feedwater nozzle inner radii and associated nozzle-to-vessel shell welds are delineated in Item Nos. B3.90 and B3.100. The ASME Code required method of examination is volumetric. With respect to the extent of examination, all nozzles with full penetration welds to the vessel shell (or closure head) and integrally cast nozzles must be examined each interval. All the Columbia nozzle-to-vessel shell welds identified in Table 1 above are full penetration welds.

3.1.4 Reason For Request

The licensee stated in its letter dated April 22, 2020, that it recognizes that the NRC intends to prohibit use of ASME Code Case N-702 for the period of extended operation in its proposed Regulatory Guide (RG) 1.147, Revision 19, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1." The licensee notes that the feedwater nozzles are excluded from ASME Code Case N-702. The licensee stated that in the proposed rulemaking for RG 1.147, Revision 19 (ADAMS Accession No. ML18099A051), the NRC states, in part:

> For the period of extended operation, the application of Code Case N-702 is prohibited. Licensees that wish to use Code Case N-702 in the period of extended operation may submit relief requests based on BWRVIP-241, Appendix A, "BWR Nozzle Radii and Nozzle-to-Vessel Welds Demonstration of Compliance with the Technical Information Requirement of the License Renewal Rule 10 CFR 54.21," approved on April 26, 2017, or plant-specific probabilistic fracture mechanics analysis.

The licensee also stated that its plant-specific analysis performed based on the BWRVIP-241-A methodology satisfies the concern of the prohibition. The licensee further stated that the proposed alternative provides an acceptable level of quality and safety based on a plant-specific evaluation using a probabilistic fracture mechanics (PFM) analysis methodology in BWRVIP-108-A and BWRVIP-241-A.

3.1.5 Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), the licensee requested relief from performing the required examinations on 100 percent of the feedwater nozzle inner radii and associated nozzle-to-shell welds as shown in Table 1 above. As an alternative, the licensee proposed to examine a minimum of 25 percent of the feedwater nozzle inner radii and a minimum of 25 percent of the associated nozzle-to-shell welds each 10-year ISI inspection interval using volumetric inspection methods performed in accordance with ASME Code, Section XI, Appendix VIII, as modified by 10 CFR 50.55a.

3.1.6 Basis for Use

The licensee stated in its letter dated April 22, 2020, that BWRVIP-108-A, BWRVIP-241-A, and ASME Code Case N-702 explicitly exclude RPV feedwater nozzles from inspection reduction because the inspection of the feedwater nozzles are managed under a separate mandated program directed by NUREG-0619, Revision 1, "BWR Feedwater Nozzle And Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10 (Technical

Report),” dated November 1980 (ADAMS Accession No. ML031600712). The licensee proposed that the feedwater nozzle radii and nozzle-to-shell welds examinations mandated under NUREG-0619 can be subsumed by the current ASME Code, Section XI requirements, and the number of inspections reduced based on this relief request.

3.1.6.1 Discussion on NUREG-0619

The licensee stated that Columbia complies with the inspection requirements of NUREG-0619 by implementation of the NRC-approved alternative General Electric (GE) report, GE-NE-523-A71-0594-A, “Alternate BWR Feedwater Nozzle Inspection Requirements, “ Revision 1, dated May 2000 (ADAMS Accession No. ML003723265), which stipulates inspections for the feedwater nozzle inner radii, nozzle inner bore regions, and the spargers. The feedwater nozzle inner radii inspections are performed in compliance with the ASME Code, Section XI, as modified by the performance demonstration initiative program and meet the established criteria of the GE report. Section 6.3, “Implementation Schedule,” of the GE report states that after compliance with ASME Code, Section XI, Appendix VIII, the examination frequency will be the ASME Code Section XI examination frequency for non-interference fit plants such as at Columbia.

The licensee noted that NRC NUREG-2221 “Technical Basis for Changes in the Subsequent License Renewal Documents NUREG-2191 and NUREG-2192,” dated December 2017 (ADAMS Accession No. ML17362A126), states that the recommendation for condition monitoring of the feedwater nozzles be performed under the ASME Code, Section XI ISIs to manage cracking in BWR feedwater nozzles induced by cyclical loading. This is based in part on improvements in the ASME Code, Section XI directed volumetric examination techniques, which now meet the requirements proposed in the GE report. The licensee stated that the feedwater nozzle examinations need no longer go beyond the ASME Code, Section XI requirements, and a separate mandated program is not required.

3.1.6.2 Justification for Reduction to 25 Percent Inspection

The licensee stated in its letter dated April 22, 2020, that using the same analytical methodology as in BWRVIP-108-A and BWRVIP-241-A, can show that failure probabilities of feedwater nozzle at the nozzle radius region and the nozzle-to-vessel shell weld due to a low temperature over pressure (LTOP) event are very low and meet the NRC acceptance criteria in NUREG-1806, “Technical Basis for Revision of Pressurized Thermal Shock (PTS) Screen Limit in the PTS Rule (10 CFR 50.61).” The licensee further stated that “[b]ased on the results of the plant-specific evaluation, and industry and internal operating experience, the inspection of 25% of the RFW nozzles is considered technically justified.”

The licensee performed a plant-specific PFM analysis using plant-specific feedwater nozzle stresses with probabilistic distributions from BWRVIP-108-A and BWRVIP-241-A to derive failure probabilities for the Columbia feedwater nozzles. The calculated failure probabilities for the feedwater nozzle and associated weld are less than the allowable failure probability in NUREG-1806.

3.1.6.3 Operating Experience

The licensee stated in its letter dated April 22, 2020, that a detailed evaluation of the historical degradation of the feedwater nozzle and sparger is presented in GE report, NEDE-21812, “BWR Feedwater Nozzle/Sparger Final Report,” March 1978. In November 1980, the NRC published NUREG-0619 to address instances of thermal fatigue cracking of the feedwater and control rod drive return line nozzle. NUREG-0619 provides guidance on design, operating, and inspection recommendations to address these concerns. The solution of the feedwater nozzle and sparger cracking problems involved several elements, including material selection and processing, nozzle clad elimination, and thermal sleeve and sparger redesign. The licensee indicated that Columbia implemented these changes during construction, including clad elimination around the feedwater nozzle and a welded thermal sleeve and safe end design.

The licensee reviewed the most recent examination results for subject components and reported that no recordable indications in the Columbia feedwater nozzle inner radii or associated nozzle-to-vessel shell welds. The licensee performed ultrasonic examinations of subject components in accordance with the ASME Code, Section XI, 2001 Edition (or later) and based on the Performance Demonstrated Initiative Program as modified by 10 CFR 50.55a. The licensee reported that all the examinations had greater than 99 percent examination coverage.

The licensee stated that a survey of all U.S. BWRs shows that the majority of RPV nozzles had no reportable indications in the nozzle inner radii or nozzle-to-shell welds. A few nozzles contained subsurface indications, which were determined to be acceptable. The survey data indicate that the inspections performed as of that date, using reliable techniques, have shown that there are no active degradation mechanisms for the RPV nozzle inner radii and nozzle-to-shell weld regions. The licensee searched the Institute of Nuclear Power Operations database in January 2020 and found that this trend appears to continue with no reports of indications in RPV nozzle-to-shell welds or inner radii regions.

3.1.7 Duration Of Proposed Alternative

The licensee requested that the duration of the proposed alternative be the remainder of plant life including the period of extended operation ending December 12, 2043.

3.2 NRC Staff Evaluation

The NRC staff evaluates whether the proposed 25 percent inspection of the feedwater nozzle assemblies will provide reasonable assurance of structural integrity of the feedwater nozzles and associated nozzle-to-shell welds at the end of plant life based on a risk-informed approach. As part of its evaluation, the NRC staff verifies whether the licensee's methodology is consistent with that of BWRVIP-108-A and BWRVIP-241-A. If differences exist between the licensee's methodology and the two BWRVIP reports, the NRC staff evaluates whether the licensee's plant-specific analysis is acceptable in terms of fracture mechanics theory.

The licensee used deterministic fracture mechanics (DFM) and PFM analyses to support the proposed alternative. To review the licensee's DFM evaluation, the NRC staff focused on the component description, finite element model, applied loading, and stresses distribution. To review the licensee's PFM evaluation, the NRC staff focused on the assumptions, input parameters, stress distribution, stress corrosion cracking (SCC), fatigue cracking, probability of detection, and results. Lastly, the NRC staff reviewed the past and future inspections of the feedwater nozzles and associated welds at Columbia.

The licensee's analysis is contained in Enclosure 2 to the April 22, 2020, letter. The NRC staff's request for additional information (RAI) and the licensee's response are contained in the licensee's letter dated October 22, 2020.

3.2.1 Background

The licensee proposed alternative is related to reducing the inspection of number of feedwater nozzles. As such, the DFM analysis is not able to evaluate the impact of the nozzle inspection reduction on the structural integrity of feedwater nozzle assemblies. The DFM analysis only considers whether the flaw will grow through wall. To assess the impact of inspection reduction on the structural integrity of the feedwater nozzle assemblies, a PFM evaluation is needed to be performed.

The PFM evaluation is an integrated probabilistic assessment of the feedwater nozzle radii and nozzle-to-shell welds to determine the frequency of events that challenge these components and the conditional failure probability of a crack penetrating through the feedwater nozzle radii and nozzle-to-shell welds. The PFM evaluation includes the frequency of events that challenge the integrity of these components and the conditional failure probability of a crack penetrating through these components because of the limiting event such as the LTOP event. The failure probability of these components is the product of the frequency of the limiting event (e.g., the LTOP event) and the conditional failure probability of a crack penetrating through the pressure boundary of these components because of the limiting event.

The PFM evaluation uses DFM equations and input parameters to calculate the applied stress intensity factor K_I and the crack growth. In the PFM evaluation, the inspection of a specific number of feedwater nozzles is factored in the probability of detection (POD) of the examination. That is, the probability of detecting a flaw is greater if all six feedwater nozzles are examined (i.e., 100 percent inspection) vs. only a few feedwater nozzles are examined (i.e., 25 percent inspection). If a flaw is detected in the feedwater nozzle based on the 25 percent inspection, then the flaw is further analyzed to determine whether it would fail by either growing to unacceptable depth size or if its applied K_I exceeds material fracture toughness, K_{Ic}.

The licensee's PFM evaluation uses the Monte Carlo simulation to randomly choose key parameters for each of the one million simulations to determine the probability of failure for the feedwater nozzle inner radii and nozzle-to-shell welds under the proposed 25 percent inspection alternative.

As part of risk-informed regulation, the NRC specified the acceptance criteria in Section 5.0 of the NRC SE for BWRVIP-108-A and BWRVIP-241-A to ensure that the probability of failure (PoF) from the supporting PFM results for RPV nozzles be less than the NRC safety goal of $5x10^{-6}$ per reactor-year as part of the NRC's risk-informed regulations. The licensee recognized that the acceptance criterion of $5x10^{-6}$ per year is the limiting factor to determine the difference in PoF per year due to the LTOP event when changing from 100 percent ISI to 25 percent inspection for the period of extended operation (i.e., 60 years). The acceptance criterion of $5x10^{-6}$ per year is taken from the NRC report, NUREG-1806, Volume 1, Chapter 10 (ADAMS Accession No. ML072830081). The licensee explained that if the resulting PoF per year due to a postulated LTOP event (including $1x10^{-3}$ probability of LTOP event occurrence per year) is estimated to be less than the allowable PoF of $5\ x10^{-6}$ per year, then there is a technical basis for the proposed inspection reduction based on the risk-informed approach.

3.2.2 Component Description

Enclosure 2, Section 4.1.1, provides the material specifications and dimensions of the reactor vessel, feedwater nozzles and nozzle-to-shell welds. The feedwater pipe material is SA-106, Grade B. The RPV material is SA-533, Grade B Class 1. The RPV cladding material is SA-

336, Grade F8 equivalent (18 Chromium – 8 Nickel). Enclosure 2, Figure 1 shows the diagram of the feedwater nozzle. The inside surface of the Columbia reactor vessel is clad with stainless weld. As stated under Operating Experience in Section 3.1.6 above, the licensee removed the cladding at the feedwater nozzles and associated attachment welds as part of modifications to minimize cracking. For the normal operating condition, the licensee assigned the pressure and temperature as 1151 pounds per square inch (psi) and 552 degrees Fahrenheit (°F), respectively.

The licensee used the material properties of metals from the 1971 Edition of the ASME Code with addenda through Summer 1971, except that heat transfer properties are from the Summer 1972 Errata.

3.2.3 Finite Element Model

The licensee used a 3-dimensional finite element model to calculate the stress distributions in the feedwater nozzle radius and nozzle-to-shell weld. The finite element model includes a portion of the reactor vessel shell, feedwater nozzle, nozzle-to-shell weld, thermal sleeve, safe end, and the nozzle-to-safe end weld.

In response to RAI 2.1, the licensee confirmed that the finite element model does not include cladding on the inner surfaces of feedwater nozzles and associated nozzle-to-shell welds, but cladding is modeled on the inside surface of the RPV base metal. The licensee explained that because the cladding was explicitly included in the model as part of vessel shell, the clad stress due to the thermal expansion difference between the cladding material and the reactor vessel shell material was accounted for in the finite element analysis. The licensee provided a sketch based on the feedwater nozzle assembly drawing, showing that the distance from the outside edge of the nozzle-to-shell weld to the edge of the cladding is greater than ¼ inch. In addition, the licensee confirmed that the finite element model of the feedwater nozzle radius and nozzle-to-shell weld was consistent with the actual field configuration. The NRC staff finds that the licensee's finite element model is consistent with the nozzle and weld configuration on the design drawing, and therefore, is acceptable.

The NRC staff noted that the licensee used a quarter size feedwater nozzle in its finite element model whereas the reactor vessel nozzle analyzed in BWRVP-108-A and BWRVIP-241-A is a complete 360-degree model. The NRC staff questioned the adequacy of the quarter size model and how the stresses were calculated in a quarter size model. In response to RAI 2.2, the licensee stated that the feedwater piping and nozzle were axisymmetric. The nozzle-to-shell weld had two axes of symmetry in the longitudinal and circumferential directions of the vessel shell. The licensee further stated that a quarter model (0 degree to 90 degrees) with the appropriate boundary conditions was adequate for the stress analysis of thermal and pressure loads that were axisymmetric and result in the same stresses as a 360-degree model. The licensee explained that for nozzle moment loadings due to thermal expansion, which were not axisymmetric, constructed a separate full 360-degree 3-dimensional finite element model from the quarter size model by reflecting about the symmetry planes. On the full 360-degree 3-D model, the licensee applied unit piping interface moments at the free end of the pipe.

The licensee explained that the stresses for the quarter size model for axisymmetric pressure and thermal loads were repeated in the other three quadrants. The licensee stated that the repeatable nature of the stress distributions can be observed in Figures 4-30 through 4-37 in BWRVIP-108-A where a 360-degree model was used. The NRC staff finds that the licensee has satisfactorily clarified that the quarter-sized finite element model in Figure 3 of Enclosure 2, provides acceptable stress distributions in the feedwater nozzle radius region and nozzle-to-shell welds for the pressure and thermal loads.

The licensee stated that for the nozzle location and crack model, the applicable stress is the stress perpendicular to a path defined 90 degrees from the tangent drawn at the blend radius. To analyze the nozzle-to-shell weld, the licensee modeled either a circumferential or an axial crack, depending on weld orientation. The licensee stated that the flaw in the weld can initiate due to either fabrication (i.e., considering only welding process) or SCC. The NRC staff finds acceptable that the licensee considered circumferential and axial cracks in the finite element model.

3.2.4 Applied Loading

The licensee considered cyclic loads such as thermal transient loads and internal pressure that would cause fatigue. Concurrent with thermal transient loads are the corresponding nozzle moment loads. Thermal transients cause expansion of the metal thereby creates bending moments on the component. The licensee derived an equation that calculated thermal expansion moment based on various feedwater and main steam temperatures. From the equation, the licensee generated two thermal expansion moment cases for thermal loading, where the feedwater temperatures were 100 °F and 420 °F, and the reactor vessel temperature was 528 °F. In addition, the licensee increased the thermal bending moment by 10 percent for possible future changes to piping analyses.

The licensee also applied a unit bending moment of 1,000 pounds-inch on the feedwater nozzle. For those feedwater nozzle or weld locations where the calculated bending moment is different from the unit bending moment, the licensee scaled the stresses derived from the unit moment loading to obtain the stresses in those locations based on the calculated moment due to thermal expansion per each transient.

The licensee derived the bounding thermal transients as shown in Enclosure 2, Table 5, by applying temperatures and the appropriate heat transfer coefficients to the inside surface of the finite element model. The licensee obtained the bounding transients for feedwater nozzle based on the load evaluation of the worst transients, which have a severe rate of temperature change and a considerable large temperature difference. The licensee stated that the transients for the feedwater nozzles and attachment welds were analyzed for 60 years of operation. In response to RAI 2.5, the licensee stated that it used plant-specific transients from the Columbia thermal cycle diagrams as shown in Tables 1 through 4 of Enclosure 2.

With respect to pressure loading, the licensee applied a unit pressure of 1000 psi with annular cap loads to the finite element model of the feedwater nozzle. The licensee used 1151 psi for the normal operating pressure at the feedwater nozzle. For the transient event, the licensee used 1200 psi and 100 °F to calculate stresses for the LTOP event. The licensee stated that the LTOP event input is consistent with the NRC's SE for BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05P)," EPRI TR-105697, dated September 1995, Proprietary.

The NRC staff questioned whether the licensee considered the forces and moments generated by the feedwater piping imposed on the feedwater nozzle. Based on the licensee's response to RAI 2.4, the NRC staff finds that the licensee has properly considered the forces and moments from the feedwater piping in calculating stress distribution in the feedwater nozzle.

In Enclosure 2, Section 3.1.2, the licensee stated that only cyclic loads such as thermal transient and pressure are included in the stress analysis, and that deadweight, which does not cycle, is not needed. The NRC noted that non-cyclic loads such as deadweight and residual stress should be included to calculate cyclic fatigue crack growth because they increase the mean stress. In response to RAI 2.6, the licensee stated that deadweight moment load was not considered because it is relatively small compared to the thermal moments and it is not cyclic. The licensee stated that deadweight only causes a mean stress in fatigue. In the fatigue analysis, the licensee used a high R-ratio of 0.7 consistent with Section 5.4 of BWRVIP-108-A to account for the increase in the mean stress. The licensee further stated that this is also consistent with the examples in Sections 4.1, 4.3 and 4.4 of BWRVIP-241-A where deadweight is not included in the analyses. The licensee indicated that weld residual stress (after post weld heat treatment) was included in the PFM design input as shown in Enclosure 2, Section 5.2.1.3, to calculate PoF. The NRC staff finds that the licensee has appropriately addressed the NRC staff's question on the deadweight and weld residual stress in the licensee's analysis. Therefore, the NRC staff finds it acceptable that the deadweight load is not included in the stress analysis.

With respect to exclusion of the seismic loads, the licensee stated that for vessels and associated nozzles, the seismic load cycles are relatively small (i.e., 50 cycles) as compared to thermal transients. Therefore, the licensee did not include seismic loads in the stress analysis. The licensee stated that this approach is consistent with the methodology in BWRVIP-108-A. The NRC staff finds that the seismic load cycles are low as compared to the thermal transients, and BWRVIP-108-A does not include the seismic loads in its evaluation. Therefore, the NRC staff finds it acceptable that the seismic load is not included in the Columbia stress analysis.

The NRC staff notes that the NRC SEs for BWRVIP-108-A and BWRVIP-241-A impose a condition that requires the maximum reactor vessel heatup/cooldown rate be limited to less than 115 ºF/hour (°F per hour). In its responses to RAI 2.7, the licensee stated that per Columbia's technical specification, the reactor vessel operational heatup/cooldown rate is limited to 100 °F/hr. The NRC staff finds that Columbia has satisfied this condition.

The NRC staff found differences between the thermal transient cycles used in the licensee's analysis as shown in Table 5 in Enclosure 2, and the thermal transient cycles used in the recirculation outlet nozzle at Columbia as shown in Table 5-5 of BWRVIP-241-A. Based on the licensee's response to RAI 2.8, the NRC staff finds that the licensee has used appropriate and bounding transients for its stress analysis. Additionally, based on the licensee's response to RAI 2.9, the NRC staff finds it acceptable that the licensee has considered appropriate loading cases in the finite element analysis of the feedwater nozzle.

3.2.5 Stresses

Enclosure 2, Section 4, discusses the stresses calculated in the feedwater nozzle and nozzle-to-shell weld based on the finite element analysis. The licensee selected six critical stress paths in the finite element model to perform PFM. The four stress paths (P1 through P4) are at nozzle-to-vessel shell blend radius and nozzle inside surface to outer blend radius, and two stress paths (P5 and P6) at the nozzle-to-vessel shell welds.

The licensee selected stress paths P1 and P2 as critical stress paths because their stress intensities are maximum at the outer nozzle blend radius based on the analysis of the unit pressure load and unit moment load analysis. The licensee selected stress paths P3 and P4 because these paths have the maximum stress intensities at the inside surface of the nozzle.

The licensee selected stress paths P5 and P6 because they correspond to the approximate center of the nozzle-to-vessel weld. For stress paths P1, P3, and P5, the licensee extracted the hoop stress S_Y. For stress paths P2, P4, and P6, the axial stress S_Z is extracted. The output of the stress analysis consists of third order polynomial stress coefficients and is used in the PFM analysis. The NRC staff finds it acceptable that the licensee selected the appropriate paths in the nozzle and weld regions because these paths experience maximum stresses that will result in bounding flaw growth.

3.2.6 Deterministic Fracture Mechanics Evaluation

The NRC staff noted that Enclosure 2 includes input parameters for a DFM evaluation, but the licensee did not discuss the results of the DFM evaluation. The NRC staff determined that the DFM evaluation should be part of the integrated evaluation to demonstrate structural integrity of the feedwater nozzle assemblies after reducing the number of feedwater nozzles inspected.

Based on the licensee's response to RAI 2.10, the NRC staff determines that the results of the licensee's DFM evaluation show that if a flaw exists and grows in a feedwater nozzle or associated weld, the structural integrity of the feedwater nozzle and weld would be maintained for 60 years. Therefore, the NRC staff finds that the licensee has demonstrated that the postulated flaw will not cause leakage or ruptures in the feedwater nozzles and attachment welds for 60 years.

3.2.7 Probabilistic Fracture Mechanics Evaluation

The licensee used the computer software VIPERNOZ to perform the PFM evaluation. The VIPERNOZ code is derived from the computer software VIPER that was used in BWRVIP-05. The VIPER code is used to analyze the shell welds in the reactor vessel whereas VIPERNOZ is specifically used to analyze the reactor vessel nozzles and associated nozzle-to-shell welds.

The VIPERNOZ code performs millions of Monte Carlo simulations using randomly selected values for the input variables to determine PoF of a component. The Monte Carlo simulation is a numerical probabilistic analysis approach which is amenable to statistical problems governed by many random variables for which closed form solutions are impossible or impractical. The essence of the simulation is to assign mean values and statistical distributions to all the key variables affecting the problem. Solution algorithms for the problem are then set up (generally in the form of a computer program) in the same manner as one would if each variable were a known, deterministic parameter. The algorithms are then exercised repetitively, many times, randomly selecting a different value for each random variable from its respective distribution for each repetition (or iteration). In PFM, each iteration results in either a failure or a non-failure (i.e., realization). The PoF of the component is simply the total number of failures divided by the total number of iterations performed.

In response to RAI 3.11, the licensee stated that for each simulated feedwater nozzle, the failure criteria within VIPERNOZ are consistent with those used in BWRVIP-108-A and BWRVIP-241-A. The NRC staff finds that the licensee's failure criteria are consistent to those of BWRVIP-108-A and BWRVIP-241-A, and therefore, are acceptable.

The licensee states that it follows the PFM methodology in BWRVIP-05, BWRVIP-108-A, and BWRVIP-241-A. In response to RAI 3.1, the licensee provided the VIPERNOZ flow diagram showing that it is similar to the flow diagram in BWRVIP-05. The licensee stated that using the VIPERNOZ code, the through-wall stresses for each stress path was input into the fracture mechanics models for the Columbia nozzle-to-shell weld or the feedwater nozzle corner crack. The licensee stated that this approach is based on Section 5.4 of BWRVIP-241-A. The NRC staff verified that the VIPERNOZ flow diagram is similar to the flow diagram in BWRVIP-05 and that the stresses are included in the models and are consistent with the method in BWRVIP-241-A.

Although the licensee's feedwater nozzle analysis was based on BWRVIP-108-A and BWRVIP-241-A, the NRC staff noted that both BWRVIP reports do not analyze feedwater nozzles from any BWR plants. In response to RAI 3.2, the licensee stated that these two BWRVIP reports are applicable to Columbia feedwater nozzles because the same Version 1.1 of VIPERNOZ was used in the Columbia PFM evaluation and in the evaluations contained in BWRVIP-108-A and BWRVIP-241-A. The NRC staff finds that VIPERNOZ used in the Columbia PFM evaluation is acceptable because its version is the same as the VIPERNOZ version used in both BWRVIP reports.

3.2.7.1 Probabilistic Fracture Mechanics Assumptions

As stated in Enclosure 2, the licensee assumed the following in the PFM evaluation for Columbia.

1. Flaws are assumed to be aligned parallel with the weld direction as justified in BWRVIP-05.

2. One stress corrosion crack initiation and 0.1 fabrication flaws are assumed per nozzle blend radius as justified in BWRVIP-108-A [SE].

3. One stress corrosion crack initiation and 1.0 fabrication flaw are assumed per nozzle-to-shell weld as justified in BWRVIP-108-A.

4. The NRC Pressure Vessel Research Users' Facility (PVRUF) flaw size distribution [is] assumed to apply.

5. The weld residual stress distribution at the nozzle-to-shell weld is assumed to be a cosine distribution through the wall thickness with 8 ksi [thousand pounds per inch] mean amplitude and 5 ksi standard deviation from BWRVIP-108-A.

6. Upper shelf fracture toughness is set to 200 ksi√in with a standard deviation of 0 ksi√in for unirradiated material, consistent with BWRVIP-108-A.

7. Standard deviation of the mean K_{Ic} is set to 15 percent of the mean value of the K_{Ic} as justified in BWRVIP-108-A.

The NRC staff finds the licensee's assumptions acceptable because they are consistent with that of BWRVIP-05 or BWRVIP-108-A.

3.2.7.2 Input Parameters

The licensee used the plant-specific feedwater nozzle dimensions and normal operating conditions, considering thermal cycles.

As stated above, the licensee assumed K_{IC} of the feedwater nozzle to be 200 ksi√in. The NRC staff noted that K_{IC} of 200 ksi√in is applicable to material at high temperature. K_{IC} at lower temperature would be lower than 200 ksi√in. The NRC staff questioned whether the temperature at the feedwater nozzle and nozzle-to-shell weld stays at sufficient high temperature at the time of maximum total applied load to qualify for the use of 200 ksi√in. In response to RAI 3.4, the licensee indicated that per Figure A-4200-1 of ASME Code, Section XI, Appendix A, the value of K_{Ic} is 220 ksi√in at the time of maximum total applied load for the Columbia feedwater nozzle. The licensee stated that the lower value of 200 ksi√in used in the PFM evaluation is conservative. Based on the licensee's clarification, the NRC staff finds acceptable that the licensee used a K_{IC} of 200 ksi√in in the PFM evaluation.

The licensee used random variables in the PFM evaluation as shown in Table 8 in Enclosure 2. Enclosure 2, Table 8 presents the chemistries (percent copper (Cu) and percent nickel (Ni)) along with the standard deviation and distributions for the feedwater nozzle forging, and the nozzle-to-vessel welds based on data from BWRVIP-108-A. The licensee stated that feedwater nozzles experience neutron fluence less than 1.00E+17 neutron/centimeter^2 (n/cm^2) for 60 years of plant operation. As such, a bounded neutron fluence of 1.00E+17 n/cm^2 is used for the evaluation of the feedwater nozzles. The NRC staff finds it acceptable that the licensee has used the material data from BWRVIP-108-A and the bounding neutron fluence for Columbia. The NRC staff further finds that the number of flaws assumed for the nozzle and nozzle-to-shell weld is consistent with BWRVIP-108-A, and therefore, is acceptable.

3.2.7.3 Stress Distribution

From the results of the finite element analysis, the licensee obtained stresses in six stress paths along the feedwater nozzle radius and nozzle-to-shell weld. These stresses are applied in the PFM evaluation to calculate flaw growth. The stresses are also used to calculate stress intensity factors, which is calculated at the deepest point of the crack.

Consistent with BWRVIP-108-A, the licensee assumed the weld residual stresses in the nozzle-to-shell welds for Paths 5 and 6. The NRC staff noted that industry operating experience has shown that some welds were repaired during construction of nuclear plants. The operating experience has further shown that the repaired welds during construction generate residual stresses that could increase the probability of flaw initiation and growth. In response to RAI 3.5, the licensee stated that as documented in Columbia's Final Safety Analysis Report (FSAR) Section 5.3.3.3, "…The shells and vessel heads were made from formed low alloy steel plates and the flanges and nozzles from low alloy steel forgings…" FSAR Section 5.3.3.3 further states that "…Postweld heat treatment of 1100°F minimum was applied to all low alloy steel welds…" The licensee stated that this heat treatment should reduce residual stresses from any repairs such that they would not be a dominant force requiring consideration in the analysis. The licensee further stated that weld residual stress (after post weld heat treatment) was included in the PFM design input. The NRC staff finds that post weld heat treatment of the attachment welds would reduce the weld residual stress. In addition, the licensee does include weld residual stress in the PFM evaluation. Therefore, the NRC staff finds that the licensee has addressed the issue of the weld residual stress in the stress analysis satisfactorily.

3.2.7.4 Stress Corrosion Cracking

For one simulation, VIPERNOZ takes stresses from Paths 1 through 6 and calculates the flaw initiation and flaw growth. Section 5.2.2.3 in Enclosure 2 of the licensee's submittal states that the SCC initiation model in the VIPERNOZ program is a power law relationship. The licensee used a lognormal distribution for the SCC initiation because it produces the best fit for the flaw initiation data. The NRC staff finds the lognormal distribution is adequate for the SCC initiation in terms of probability calculation and is, therefore, acceptable.

The licensee indicated that the SCC growth model in the VIPERNOZ program is a power law relationship taken from NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment," dated February 2007 (ADAMS Accession No. ML070710257). The licensee stated that it used a Weibull distribution for the SCC growth distribution because it produces the best fit for the SCC data. The NRC staff finds that the Weibull distribution is adequate for the SCC growth calculation because it provides an acceptable fit for the SCC crack growth data.

Section 5.2.2.3 in Enclosure 2 states that the SCC initiation data were based on cast stainless steel, which were also used in BWRVIP- 05, BWRVIP-108-A and BWRVIP-241-A. The NRC staff questioned why the SCC initiation data for cast stainless steel were used to calculate the crack growth even though the feedwater nozzle and nozzle-to-shell weld were made of low alloy steel. In response to RAI 3.6, the licensee explained that the SCC initiation values used were selected to be consistent with the NRC SE approving BWRVIP-108-A. The licensee stated that BWRVIP-05 notes that low alloy steel components such as Columbia's feedwater nozzles are highly resistant to SCC initiation in BWR environments so use of the cladding initiation coefficient is considered conservative. The NRC staff finds it acceptable that the crack initiation data for cast stainless steel were used on the low alloy steel of feedwater nozzle and weld at Columbia because operating experience has shown that low alloy steel does have resistance to SCC initiation.

3.2.7.5 Fatigue Cracking

The licensee stated that the fatigue crack growth data for SA-533 Grade B Class 1 and SA-508 Class 2 (carbon-molybdenum steels) were based on weld metal testing in reactor water environments at an R ratio (algebraic ratio of K_{min}/K_{max}) of 0.2 and 0.7. To produce a fatigue crack growth law and distribution for Columbia, the licensee fitted the data for R = 0.7 into the form of Paris Law. The licensee stated that it chose R = 0.7 fatigue crack growth law for conservatism.

Enclosure 2, Section 5.2.2.5 discusses the comparison between the fatigue crack growth data that were used in the Columbia analysis and the fatigue crack growth law in ASME Code, Section XI in a reactor water environment. The licensee stated that its fatigue crack growth data show a reasonable comparison; however, the fatigue growth law in ASME Code, Section XI, is more conservative than fatigue growth rate data used in the Columbia evaluation at high difference in stress intensity factor, ΔK ($K_{max} - K_{min}$). The NRC staff questioned whether the fatigue crack growth curves used in the Columbia PFM evaluation is adequate. Based on its response to RAI 3.7, the NRC staff finds that the fatigue growth law used in the Columbia evaluation is consistent with that of BWRVIP-108-A and BWRVP-241-A; therefore, the fatigue growth law used for Columbia is acceptable.

In response to RAI 3.3, the licensee explained that SCC crack growth and fatigue crack growth are calculated independently and summed in succession. At each time step of the Monte Carlo simulation, SCC crack growth is calculated and added first, and then, fatigue crack growth is added to the flaw size after SCC crack growth. The licensee stated that within the PFM methodology, flaw detection and vessel failure are checked after crack growth from both SCC and fatigue as can be seen in the PFM flow diagram. The NRC staff finds it acceptable that the SCC growth and fatigue crack growth are combined to calculate the flaw size because the addition will provide a conservative flaw size.

3.2.7.6 Probability of Detection

Enclosure 2, Table 10 shows cumulative POD with respect to the flaw sizes from 0.00 to 0.60 inches based on ultrasonic testing. The POD is applied to both the feedwater nozzle radius and associated weld. The NRC staff noted that the nozzle radius and associated weld are of different shape and thickness. The ultrasonic examination coverage may be different between the feedwater nozzle radius and the associated weld such that the POD may be different for the feedwater nozzle and associated nozzle-to-shell weld. In response to RAI 3.8, the licensee stated that the POD values used are identical to the POD values used in BWRVIP-108-A, Figure 2-1. The NRC staff finds it acceptable that the licensee used the same POD for the nozzle and weld because this approach is consistent with the NRC staff approved BWRVIP-108-A.

In response to RAI 3.9, the licensee explained that there is no correlation between the percentage of inspection and the POD curve. The POD curve characterizes the effectiveness of the inspection method. The licensee stated that the POD curve and inspection percentage are two separate, independent inputs into VIPERNOZ. The licensee further stated that the PFM methodology in VIPERNOZ will first consider whether a nozzle is inspected (inspection percentage) and then whether a flaw is detected (POD curve) at each time step in the simulation. The NRC staff finds that the licensee has satisfactorily clarified the POD and 25 percent inspection in the PFM evaluation.

The NRC staff noted that with the proposed inspection of 25 percent of the six feedwater nozzles, some feedwater nozzles and attachment welds may not be inspected for the remainder of the plant life. In response to RAI 3.10, the licensee responded that in Section 5.7 of BWRVIP-108-A, PFM evaluations were performed with inspection sampling of 0 percent, 25 percent, and 99 percent of various reactor vessel nozzles. In all three cases, the PoF are below the acceptance criteria. The licensee stated that considering the low PoF values for the 25 percent sample in Enclosure 2 (on the order of $1x10^{-11}$) of the licensee's submittal, similar conclusions will be reached for the 0 percent inspection. The NRC staff finds that based on the results of BWRVIP-108-A, if a feedwater nozzle was not examined for the remainder of the plant life, the PoF of the nozzle would still be within the acceptance criteria. The NRC staff finds that in terms of the theoretical prediction, if a feedwater nozzle is not examined for the remainder of the plant life, the structural integrity of that nozzle would most likely not be challenged. However, this finding does not imply that the NRC would accept 0 percent inspection for all six feedwater nozzles because 0 percent inspection does not provide reasonable assurance of structural integrity of all feedwater nozzles. This is because under the risk-informed approach, the NRC regulation is based on the deterministic evaluation with input from the probabilistic analysis to assess the structural integrity of the safety-related component to ensure public health and safety. The deterministic evaluation includes defense-in-depth measures such that some inspections will be performed on the feedwater nozzles.

3.2.7.7 Results of Probability of Failure

Enclosure 2, Section 6 provides the results of PoF for the 25 percent inspection of the feedwater nozzles and attachment welds. For normal operation, the licensee did not find that any failures occurred in one million simulations for 60 years based on 25 percent inspection of six feedwater nozzles and attachment weld. However, for the purpose of the PFM analysis, the licensee assumed one failure occurred. As such, the licensee estimated that the PoF for normal operation is less than 1.67×10^{-8} per year (one failure/one million simulations/60 years). The calculated PoF for normal operation is less than the allowable PoF of 5×10^{-6} per year and meets the acceptance criterion from NUREG-1806.

For LTOP events, the licensee's PFM evaluation indicates that for stress Path 5 at the nozzle-to-shell weld, three failures occurred in one million simulations for 60 years with 25 percent inspection. The conditional PoF for LTOP events for Path 5 is, therefore, 5.0×10^{-8} per year (three failures /one million simulations/60 years). Accounting for an LTOP event occurrence of 1×10^{-3} per year, the calculated PoF for LTOP events for Path 5 is 5.0×10^{-11} per year ($1 \times 10^{-3} \times 5.0 \times 10^{-8}$), which is less than the allowable PoF of 5×10^{-6} per year and meets the acceptance criterion from NUREG-1806.

For LTOP events for all other stress paths (i.e., Paths 1 through 4 and 6) at the nozzle blend radius and the nozzle-to-shell weld, no failures occurred in one million simulations for 60 years with 25 percent inspection. Assuming one failure did occur, the PoF is 1.67×10^{-8} per year. For all stress paths other than Path 5, multiplying an LTOP event occurrence of 1×10^{-3} per year to 1.67×10^{-8} per year, the calculated PoF for LTOP events is less than 1.67×10^{-11} per year. This is less than the allowable PoF of 5×10^{-6} per year and meets the acceptance criterion from NUREG-1806.

The NRC staff noted that the above PoF results for the Columbia feedwater nozzles or the nozzle-to-shell welds at 25 percent inspection in 60 years are much less than the PoF estimated for the recirculation nozzles for a 25 percent inspection in 40 years in BWRVIP-108-A and BWRVIP-241-A. The NRC staff further noted that in general, the feedwater nozzles should have experienced more transients than the recirculation nozzles. The Columbia nozzles were analyzed for 20 more years than the recirculation nozzles. In addition, the Columbia feedwater nozzles have more applicable transients than the nozzles analyzed in the BWRVIP-108-A supplement and BWRVIP-241-A. Therefore, it appears that the PoF for the Columbia feedwater nozzle should be higher, not lower, than the PoF for the recirculation nozzle analyzed in the BWRVIP-108-A supplement and BWRVIP-241-A. In response to RAI 3.12, the licensee explained that the PoF for the Columbia recirculation outlet nozzle reported in Table 5-9 of BWRVIP-241-A are the conditional PoF. To calculate the PoF due to an LTOP event, the BWRVIP-241-A conditional probabilities should be multiplied by the probability of an LTOP event occurrence (1×10^{-3}). The licensee also explained that the slightly higher PoF of the recirculation outlet nozzle compared to the Columbia feedwater nozzle is attributed to the conservative combination of the transients and associated cycles in the BWRVIP-241-A evaluation of the recirculation outlet nozzle. The NRC staff finds that the licensee has appropriately clarified the difference in the PoF between the Columbia feedwater nozzle and recirculation nozzles in BWRVIP-241-A.

3.2.8 Inspection

The NRC staff questioned the expansion inspections if an indication is detected in a feedwater nozzle or in a nozzle-to-shell weld. In response to RAI 1.1, the licensee stated that the proposed alternative does not seek relief from any other aspect of the ASME Code, Section XI. The licensee stated that if an indication is detected in a feedwater nozzle or an attachment weld that exceeds ASME inspection criteria, scope expansion (extent of condition) will be performed in accordance with the ASME Code, Section XI, Subsection IWB-2430, "Additional Examinations," for the Code of Record in place at the time of discovery.

The proposed alternative states that the most recent examination results reported that no recordable indications in the feedwater nozzle inner radii or nozzle-to-shell welds. In response to RAI 1.2, the licensee provided the examination coverage data showing that all six-feedwater nozzle inner radii and all six nozzle-to-shell welds have been inspected and have achieved more than 99 percent coverage based on ultrasonic testing. The NRC staff finds acceptable that the licensee has followed the inspection requirements of the ASME Code, Section XI, in the previous examination campaign, and that the Columbia feedwater nozzles and associated welds do not contain recordable indications.

3.2.9 Summary

The NRC finds that the licensee has used an appropriate DFM evaluation to demonstrate that if a flaw exists and grows in a feedwater nozzle or associated weld that is not being inspected for 60 years, the feedwater nozzle assembly would still maintain its structural integrity. In addition, the licensee has used the PFM evaluation to demonstrate that with a 25 percent inspection of the six feedwater nozzle assemblies, considering a postulated flaw with the flaw growing, the PoF for the feedwater nozzle assemblies will be within the allowable PoF for 60 years.

Based on the DFM and PFM evaluation results, along with the Columbia inspection results showing no indications of inservice degradation, the NRC staff determines that based on the risk-informed approach, the licensee has demonstrated the reasonable assurance of structural integrity of the feedwater nozzle inner radii and associated welds considering the proposed inspection of 25 percent of the six feedwater nozzles during each 10-year ISI interval until the end of plant life scheduled for December 12, 2043.

4.0 CONCLUSION

Based on information in the licensee’s letter dated April 22, 2020, as supplemented by letter dated October 22, 2020, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity of the subject feedwater nozzle inner radii and associated nozzle-to-shell welds and, therefore, provides an acceptable level of quality and safety. Accordingly, as set forth above, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC authorizes the use of Relief Request 4ISI-09 at Columbia Generating Station for the remainder of plant life including the period of extended operation ending on December 12, 2043.

All other requirements in the ASME Code, Section XI, for which relief was not specifically requested and approved in this proposed alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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Date: April 14, 2021

SUBJECT: COLUMBIA GENERATING STATION – APPROVAL FOR RELIEF REQUEST 4ISI-09, REGARDING ALTERNATE EXAMINATION OF FEEDWATER NOZZLES (EPID L-2020-LLR-0068) DATED APRIL 14, 2021

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