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

February 13, 2015

Mr. Mark E. Reddemann
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION – REQUEST FOR ALTERNATIVE 3ISI-14 TO THE REQUIREMENTS OF THE ASME CODE (TAC NO. MF3435)

Dear Mr. Reddemann:

By letter dated February 6, 2014, as supplemented by letter dated August 27, 2014, Energy Northwest (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to certain American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI requirements at Columbia Generating Station (CGS). The proposed alternative in Relief Request 3ISI-14 would revise the inspection requirements for certain reactor pressure vessel (RPV) nozzle-to-vessel welds and nozzle inner radii from those based on ASME Code, Section XI, Subarticle IWB-2500, to an alternative based on ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds."

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. The paragraph headings in 10 CFR 50.55a were changed by *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014 (e.g., 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2)). See the cross-reference tables, which are cited in the notice, in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML14015A191 and package Accession No. ML14211A050.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that Energy Northwest has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC authorizes the licensee's proposed alternative for inspection of the nozzle-to-vessel shell welds and nozzle inner radii sections of RPV nozzles listed in Section 3.1 of the enclosed safety evaluation for the duration of the third 10-year inservice inspection interval at CGS ending December 12, 2015.

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

If you have any questions regarding this matter, please contact Andrea George, the CGS Project Manager, at (301) 415-1081 or via e-mail at andrea.george@nrc.gov.

Sincerely,

Eric R. Oesterle, Acting Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE 3ISI-14 REGARDING INSPECTION REQUIREMENTS

FOR REACTOR VESSEL NOZZLE WELDS AND NOZZLE INNER RADII

COLUMBIA GENERATING STATION

ENERGY NORTHWEST

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated February 6, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14042A160), as supplemented by letter dated August 27, 2014 (ADAMS Accession No. ML14254A401), Energy Northwest (the licensee) requested an alternative to the inspection requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) for reactor pressure vessel (RPV) nozzle-to-vessel welds and nozzle inner radii to apply ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1," dated February 20, 2004. The duration of the alternative would be for the third 10-year inservice inspection (ISI) interval for Columbia Generating Station (Columbia).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the proposed alternative would apply ASME Code Case N-702, which requires a minimum of 25 percent of the RPV nozzle-to-vessel welds and inner radius sections be inspected during each 10-year ISI interval, including at least one nozzle from each type and nominal pipe size, in lieu of the 100 percent inspection requirements of ASME Code, Section XI, Subarticle IWB-2500.

The paragraph headings in 10 CFR 50.55a were changed by *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014 (e.g., 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2)). See the cross-reference tables, which are cited in the notice, at ADAMS Accession No. ML14015A191 and ADAMS package Accession No. ML14211A050.

2.0 REGULATORY EVALUATION

In accordance with 10 CFR 50.55a(g)(4), the licensee is required to perform ISI of ASME Code Class 1, 2, and 3 components and system pressure tests during the first 10-year interval and

subsequent 10-year intervals that comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(a)(1), subject to the limitations and modifications listed therein.

The regulation in 10 CFR 50.55a(z) states, in part, that the Director of the Office of Nuclear Reactor Regulation may authorize an alternative to the requirements of 10 CFR 50.55a(b)-(h). There are two justifications for an alternative to be authorized. First, per 10 CFR 50.55a(z)(1), the licensee must demonstrate that the proposed alternative would provide an acceptable level of quality and safety. For the second possible justification for an alternative to be authorized, described in 10 CFR 50.55a(z)(2), the licensee must show that following the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.0 TECHNICAL EVALUATION

3.1 Background

For all RPV nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI requires 100 percent inspection during each 10-year ISI interval. However, ASME Code Case N-702 proposes an alternative which reduces the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radii from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval.

The NRC approved the Electric Power Research Institute (EPRI) BWR Vessel Internals Project (BWRVIP)-108 report, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii,"[1] dated October 2002, in a safety evaluation (SE) dated December 19, 2007 (ADAMS Accession No. ML073600374). The NRC also approved the EPRI BWRVIP-241 report, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,"[2] dated October 2010, which contains the technical basis supporting ASME Code Case N-702, in an SE dated April 19, 2013 (ADAMS Accession No. ML13071A240). The BWRVIP-241 report contains additional probabilistic fracture mechanics (PFM) results supporting revision of the evaluation criteria in the BWRVIP-108 report. Hence, the conditions and limitations specified in the NRC staff SE for the BWRVIP-241 report supersede those conditions and limitations in the NRC staff SE for the BWRVIP-108 report.

The NRC staff's SE for the BWRVIP-241 report specified plant-specific requirements which must be met for applicants proposing to use this alternative. The licensee's request for alternative 3ISI-14 intended to demonstrate that the relevant Columbia RPV nozzle-to-vessel welds and the inner radii meet these plant-specific requirements. Licensees can demonstrate the plant-specific applicability of the BWRVIP-241 report by meeting the following general and nozzle-specific criteria specified in the NRC staff SE for the BWRVIP-241 report:

(1) The maximum RPV heatup/cooldown rate is limited to less than 115 °F/hour;

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[1] This report is publicly available from the EPRI website at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001003557.

[2] This report can be requested via the EPRI website at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001021005.

For recirculation inlet nozzles

(2) $(pr/t)/C_{RPV} \leq 1.15$

p = RPV normal operating pressure (psi),
r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
C_{RPV} = 19332;

(3) $[p(r_o^2 + r_i^2)/ (r_o^2 - r_i^2)]/C_{NOZZLE} \leq 1.47$

p = RPV normal operating pressure (psi),
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
C_{NOZZLE} = 1637;

For recirculation outlet nozzles

(4) $(pr/t)/C_{RPV} \leq 1.15$

p = RPV normal operating pressure (psi),
r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
C_{RPV} = 16171; and

(5) $[p(r_o^2 + r_i^2)/ (r_o^2 - r_i^2)]/C_{NOZZLE} \leq 1.59$

p = RPV normal operating pressure (psi),
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
C_{NOZZLE} = 1977.

The BWRVIP-241 report documents additional PFM results supporting revision of the five evaluation criteria in the BWRVIP-108 report. Since the objective of the BWRVIP-241 report is limited (i.e., revision of the limitations and conditions specified in the December 19, 2007, SE for the BWRVIP-108 report), it is considered as a supplement to the BWRVIP-108 report, not a replacement. Nonetheless, the conditions and limitations specified in the SE for the BWRVIP-241 report supersede those in the SE for the BWRVIP-108 report.

The BWRVIP-108 NRC staff SE concluded, in part, that (1) the fracture toughness-related reference temperature (RT_{NDT}) used in the PFM analyses were based on data from the entire fleet of BWR RPVs, making the PFM analyses bounding with respect to fracture resistance and leaving the driving force of the underlying PFM analyses the only item to be evaluated, and (2) except for the RPV heatup/cooldown rate, the plant-specific criteria are for the recirculation inlet and outlet nozzles only because the probabilities of failure, P(F|E)s, for other nozzles are an order of magnitude lower. Based on the above, the BWRVIP-241 report documents additional PFM analyses on the recirculation inlet and outlet nozzles having the highest driving

force among the BWR fleet to demonstrate that the associated vessel failure probability for the normal operation is still consistent with the NRC safety goal, thus supporting the proposed revision of the five evaluation criteria. The SE for the BWRVIP-241 report accepted the proposed revision of the five evaluation criteria in the BWRVIP-108 report.

3.2 The Licensee's Request for Alternative

ASME Code Class 1 Components Affected

Table 1

Identification Number	Description	Code Category	Item Number
N1-0	RRC NOZZLE TO VESSEL WELD @ 0 Deg	B-D	B3.90
N1-0-IR	RRC NOZZLE INNER RADIUS @ 0 Deg	B-D	B3.100
N1-180	RRC NOZZLE TO VESSEL WELD @ 180 Deg	B-D	B3.90
N1-180-IR	RRC NOZZLE INNER RADIUS @ 180 Deg	B-D	B3.100
N2-30	RRC NOZZLE TO VESSEL WELD @ 30 Deg	B-D	B3.90
N2-30-IR	RRC NOZZLE INNER RADIUS @ 30 Deg	B-D	B3.100
N2-60	RRC NOZZLE TO VESSEL WELD @ 60 Deg	B-D	B3.90
N2-60-IR	RRC NOZZLE INNER RADIUS @ 60 Deg	B-D	B3.100
N2-90	RRC NOZZLE TO VESSEL WELD @ 90 Deg	B-D	B3.90
N2-90-IR	RRC NOZZLE INNER RADIUS @ 90 Deg	B-D	B3.100
N2-120	RRC NOZZLE TO VESSEL WELD @ 120 Deg	B-D	B3.90
N2-120-IR	RRC NOZZLE INNER RADIUS @ 120 Deg	B-D	B3.100
N2-150	RRC NOZZLE TO VESSEL WELD @ 150 Deg	B-D	B3.90
N2-150-IR	RRC NOZZLE INNER RADIUS @ 150 Deg	B-D	B3.100
N2-210	RRC NOZZLE TO VESSEL WELD @ 210 Deg	B-D	B3.90
N2-210-IR	RRC NOZZLE INNER RADIUS @ 210 Deg	B-D	B3.100
N2-240	RRC NOZZLE TO VESSEL WELD @ 240 Deg	B-D	B3.90
N2-240-IR	RRC NOZZLE INNER RADIUS @ 240 Deg	B-D	B3.100
N2-270	RRC NOZZLE TO VESSEL WELD @ 270 Deg	B-D	B3.90
N2-270-IR	RRC NOZZLE INNER RADIUS @ 270 Deg	B-D	B3.100
N2-300	RRC NOZZLE TO VESSEL WELD @ 300 Deg	B-D	B3.90
N2-300-IR	RRC NOZZLE INNER RADIUS @ 300 Deg	B-D	B3.100
N2-330	RRC NOZZLE TO VESSEL WELD @ 330 Deg	B-D	B3.90
N2-330-IR	RRC NOZZLE INNER RADIUS @ 330 Deg	B-D	B3.100

RRC: Reactor Recirculation
Code Category B-D: "Full Penetration Welded Nozzles in Vessels"
Item Number B3.90: "Nozzle-to-Vessel Welds"
Item Number B3.100: "Nozzle Inside Radius Section"

Applicable ASME Code Edition and Addenda (as stated by the licensee)

Columbia Generating Station (Columbia) ASME Section XI Code is the 2001 Edition through the 2003 Addenda (Reference 3 [of the licensee's submittal]). Additionally, for ultrasonic examinations, Section XI, Appendix VIII, "Performance

Demonstration for Ultrasonic Examination Systems," is implemented as required (and modified) by 10 CFR 50.55a(b)(2)(xv).

ASME Code Requirement for which Alternative is Requested (as stated by the licensee)

Class 1 nozzle-to-vessel weld and nozzle inner radii examination requirements are given in Subsection IWB, Table IWB-2500-1, "Examination Category B-D Full Penetration Welds of Nozzles in Vessels - Inspection Program B," Item Numbers B3.90 and B3.100, respectively. The method of examination is volumetric. For the extent of examination, all nozzles with full penetration weld to the vessel shell (or head) and integrally cast nozzles must be examined each interval.

Licensee's Proposed Alternative and Basis for Use (as stated by the licensee)

Pursuant to 10 CFR 50.55a(a)(3)(i) [retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014], relief is requested from performing the required examinations on 100% of the identified nozzle assemblies in Table 1[3] above. As an alternative, for all welds and inner radii identified in Table 1, Columbia proposes to examine a minimum of 25% of the nozzle-to-vessel welds and inner radius sections, including at least one nozzle from each type and nominal pipe size, in accordance with Code Case N-702 (Reference 4). The proposed schedule to complete the minimum of 25% from each group is presented in Table 2[4] below. Columbia has completed three assembly (nozzle and inner radii) examinations already this interval using automated ultrasonic testing techniques. Greater than 97% coverage was achieved on each location and no relevant indications were reported. With the acceptance of the relief request only one nozzle-to-vessel weld and one inner radius would be required to meet the 25% for the current interval.

[The BWRVIP-241 report] documents supplemental analyses for BWR reactor pressure vessel (RPV) recirculation inlet and outlet nozzle-to-shell welds and nozzle inner radii. BWRVIP-241 was submitted to address the limitations and conditions specified in the December 19, 2007 [, SE for the BWRVIP-108 report]. The BWRVIP-108 report contains the technical basis supporting [ASME Code Case N-702] for reducing the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radius areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. Based on the two evaluations (BWRVIP-241 and BWRVIP-108NP), the failure probabilities due to a low temperature over pressure (LTOP) event at the nozzle blend radius region and the nozzle-to-vessel shell weld for Columbia recirculation nozzles are very low and meet the NRC safety goal.

Based on the results of this evaluation, the report concluded that the inspection of 25% of each nozzle type is technically justified as per Code Case N-702.

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[3] Table 1 of the licensee's application is provided in the first subheading of Section 3.2 of this SE.
[4] This refers to Table 2 in Section 5 of the licensee's application. Table 2 is not included in this SE.

The licensee's application continues by demonstrating its compliance with the five limitations and conditions from the NRC staff's SE for the BWRVIP-241 report. The licensee's information regarding the five conditions and limitations is not reproduced in this SE.

Duration of Alternative (as stated by the licensee)

> The duration of this request is for the third 10-year [ISI] interval ending December 12, 2015.

3.3 NRC Staff Evaluation

In its application, the licensee provided Columbia's plant-specific RPV data and its evaluation of the five driving force factors, or ratios, against the criteria established in the NRC staff SE for the EPRI BWRVIP-214 report. The NRC staff verified the licensee's analysis against the conditions and limitations in the NRC staff SE for the EPRI BWRVIP-214 report, and concludes that all the conditions and limitations for the use of EPRI BWRVIP-214 are satisfied. As a result, the reduced inspection requirements in accordance with ASME Code Case N-702 apply to all proposed Columbia RPV nozzles. Based on information provided by the licensee, and the evaluation above, the NRC staff concludes that the licensee's proposed alternative for all Columbia RPV nozzles included in this application (see Section 3.2 of this SE) provides an acceptable level of quality and safety because the plant-specific results met the revised criteria in BWRVIP-241, which are consistent with the NRC safety goal on PFM results.

It should be noted that RPV feedwater nozzles and control rod drive return line nozzles are outside the scope of ASME Code Case N-702 and are, accordingly, outside the scope of this application.

ASME Code Case N-702 stipulates that a VT-1 examination may be used in lieu of the volumetric examination for the inner radii (i.e., Item No. B3.100, "Nozzle Inside Radius Section," see Table 1 in Section 3.2 of this SE). At the time of the NRC staff review, this was not consistent with the NRC position for VT-1 examination that was established in RG 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," October 2010 (ADAMS Accession No. ML101800536), regarding ASME Code Case N-648-1, "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles." In a request for information (RAI) dated July 31, 2014 (ADAMS Accession No. ML14206A009) (EVIB-RAI-1), the NRC staff requested that the licensee provide information specifying an examination methodology in accordance with the conditions placed upon the use of ASME Code Case N-648-1. In its RAI response dated August 27, 2014, the licensees provided a revised Relief Request 3ISI-14, which indicated that Energy Northwest will perform ultrasonic testing (UT) examination of the applicable nozzle inner radius sections and is not requesting use of the VT-1 examination provisions included in the code case. In October 2014, the NRC issued RG 1.147, Revision 17 (ADAMS Accession No. ML13339A689), which updated the conditions placed on the use of ASME Code Case N-648-1. The NRC staff has reviewed the updated conditions, and concludes that the licensee's use of UT examination of the RPV nozzle inner radii is still consistent with the updated RG 1.147, Revision 17. Therefore, the NRC staff concludes that that EVIB-RAI-1 is resolved, and that the licensee's use of UT examination for the RPV nozzle inner radii is acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the request for alternative, as supplemented, regarding the licensee's evaluation of the five plant-specific criteria specified in the April 19, 2013, NRC staff SE for the BWRVIP-241 report, which provides technical bases for use of ASME Code Case N-702 to examine RPV nozzle-to-vessel welds and nozzle inner radii at Columbia. Based on the evaluation in this SE, the NRC staff concludes that the licensee's proposed alternative provides an acceptable level of quality and safety because the licensee has met the conditions required for ASME Code Case N-702 to be applicable at Columbia. As discussed in Section 3.3 of this SE, this request for alternative does not include feedwater nozzles and control rod drive return line nozzles.

Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) and is in compliance with the ASME Code's requirements. Therefore, the NRC authorizes the licensee's proposed alternative for inspection of nozzle-to-vessel shell welds and nozzle inner radii sections of RPV nozzles listed in Section 3.2 of this SE for the third 10-year ISI interval at Columbia.

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

Principal Contributor: Simon Sheng, NRR

Date: February 13, 2015

If you have any questions regarding this matter, please contact Andrea George, the CGS Project Manager, at (301) 415-1081 or via e-mail at andrea.george@nrc.gov.

Sincerely,

/RA/

Eric R. Oesterle, Acting Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

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