Alex L Javorik
Vice President, Engineering
P.O. Box 968, Mail Drop PE20
Richland, WA 99352-0968
Ph. 509-377-8555 F. 509-377-4317
aljavorik@energy-northwest.com

February 4, 2016
GO2-16-008

10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397; FOURTH TEN-YEAR INTERVAL INSERVICE INSPECTION (ISI) PROGRAM RELIEF REQUEST 4ISI-04**

References: (1) Letter dated April 19, 2013, Sher Bahadur (NRC) to Dennis Madison (BWRVIP), "Final Safety Evaluations of the Boiling Water Reactor Vessel Internals Project (BWRVIP-241) Report, ‘Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii"

(2) Letter dated December 19, 2007, Matthew A. Mitchell (NRC), to Rick Libra (BWRVIP), "Safety Evaluation of Proprietary EPRI Report, 'BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)”

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(z)(1) Energy Northwest hereby requests NRC approval of the proposed alternate to American Society of Mechanical Engineers (ASME) Section XI, Sub Article IWB-2500 to allow reduced percentage requirements for nozzle to vessel weld and inner radius examinations while still providing an acceptable level of quality and safety. This alternative is requested for the fourth ten-year interval ISI program at Columbia Generating Station. The details of the 10 CFR 50.55a request are provided as Attachment 1.

Approval of Relief Request 4ISI-04 will allow reduced examination requirements through application of ASME Code Case N-702. The applicability of Code Case N-702 to Columbia Generating Station has been demonstrated by meeting the criteria in Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 (Reference 1) as shown in Attachment 1.

Energy Northwest requests approval by February 9, 2017 to accommodate application of the request during the next refueling outage.

There are no new commitments made in this submittal. If you have any questions or require additional information, please contact Lisa Williams at 509-377-8148.

Executed this 3rd day of February, 2016.

Respectfully,

A. L. Javorik
Vice President, Engineering

Attachment: As Stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda - BPN1399 (email)
WA Horin - Winston & Strawn
RR Cowley -WDOH (email)
EFSECutc.wa.gov-- EFSEC (email)

10 CFR 50.55a Relief Request Number 4ISI-04

Alternative Requirements for Nozzle Inner Radius and Nozzle-to-Shell Welds

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(1)

--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

Code Class: 1

Reference: American Society of Mechanical Engineers (ASME) Section XI, Table IWB-2500-1

Examination Category: B-D

Item Number: B3.90 and B3.100

Component Numbers: Reactor Pressure Vessel (RPV) Nozzles: N1, N2, N3, N5, N6, N7, N8, N9, N16, and N18

The components in Table 1 are affected by this request.

| **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 |

| Table 1 | | | |
|---|---|---|---|
| **Identification Number** | **Description** | **Code Category** | **Item Number** |
| 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 |
| | | | |
| N3-72 | MS Nozzle to Vessel Weld @ 72 Deg | B-D | B3.90 |
| N3-72-IR | MS Nozzle Inner Radius @ 72 Deg | B-D | B3.100 |
| N3-108 | MS Nozzle to Vessel Weld @ 108 Deg | B-D | B3.90 |
| N3-108-IR | MS Nozzle Inner Radius @ 108 Deg | B-D | B3.100 |
| N3-252 | MS Nozzle to Vessel Weld @ 252 Deg | B-D | B3.90 |
| N3-252-IR | MS Nozzle Inner Radius @ 252 Deg | B-D | B3.100 |
| N3-288 | MS Nozzle to Vessel Weld @ 288 Deg | B-D | B3.90 |
| N3-288-IR | MS Nozzle Inner Radius @ 288 Deg | B-D | B3.100 |
| | | | |
| N5-120 | LPCS Nozzle to Vessel Weld @ 120 Deg | B-D | B3.90 |
| N5-120-IR | LPCS Nozzle Inner Radius @120 Deg | B-D | B3.100 |
| | | | |
| N6-45 | LPCI Nozzle to Vessel Weld @ 45 Deg | B-D | B3.90 |
| N6-45-IR | LPCI Nozzle Inner Radius @ 45 Deg | B-D | B3.100 |
| N6-135 | LPCI Nozzle to Vessel Weld @ 135 Deg | B-D | B3.90 |
| N6-135-IR | LPCI Nozzle Inner Radius @135 Deg | B-D | B3.100 |
| N6-315 | LPCI Nozzle to Vessel Weld @ 315 Deg | B-D | B3.90 |
| N6-315-IR | LPCI Nozzle Inner Radius @ 315 Deg | B-D | B3.100 |
| | | | |
| N9-105 | JP Instrumentation Nozzle to Vessel Weld @ 105 Deg | B-D | B3.90 |
| N9-105-IR | JP Instrumentation Nozzle Inner Radius@ 105 Deg | B-D | B3.100 |
| N9-285 | JP Instrumentation Nozzle to Vessel Weld @ 285 Deg | B-D | B3.90 |
| N9-285-IR | JP Instrumentation Nozzle Inner Radius@ 285 Deg | B-D | B3.100 |
| | | | |
| N16-240 | HPCS Nozzle to Vessel Weld @ 240 Deg | B-D | B3.90 |
| N16-240-IR | HPCS Nozzle Inner Radius @ 240 Deg | B-D | B3.100 |
| | | | |
| N7 | Top Head Spray Nozzle to Top Head Weld | B-D | B3.90 |
| N7-IR | Top Head Spray Nozzle Inner Radius | B-D | B3.100 |

**Table 1**

| Identification Number | Description | Code Category | Item Number |
|---|---|---|---|
| | | | |
| N8 | Top Head Vent Nozzle to Top Head Weld | B-D | B3.90 |
| N8-IR | Top Head Vent Nozzle Inner Radius | B-D | B3.100 |
| | | | |
| N18 | Top Head Spare Nozzle to Top Head Weld | B-D | B3.90 |
| N18-IR | Top Head Spare Nozzle Inner Radius | B-D | B3.100 |
| Reactor Recirculation (RRC)<br>Jet Pump (JP)<br>Low Pressure Core Spray (LPCS)<br>Low Pressure Core Injection (LPCI)<br>High Pressure Core Spray (HPCS)<br>Main Steam (MS) | | | |

2. Applicable Code Edition and Addenda

The applicable ASME Section XI Code Edition and Addenda for Columbia Generating Station's (Columbia) fourth ten-year ISI interval is the 2007 Edition through the 2008 Addenda. Additionally, for ultrasonic examinations, Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," is implemented as required and as modified by 10 CFR 50.55a.

3. Applicable Code Requirement

The applicable Code requirement is contained in Subsection IWB, Table IWB-2500-1, "Examination Category B-D, Full Penetration Welded Nozzles in Vessels." Class 1 nozzle-to-vessel weld and nozzle inner radii examination requirements are delineated in Item Number B3.90 "Nozzle-to-Vessel Welds," and B3.100, "Nozzle Inside Radius Section." The method of examination is volumetric. With respect to the extent of examination, all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles must be examined each interval. All of the nozzle assemblies identified in Table 1 are full penetration welds.

4. Reason for Request

The Federal Register Notice (FRN) published November 5, 2014, contains the rulemaking that amends 10 CFR 50.55a to incorporate by reference Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." As stated in the FRN, licensees may use the Code Cases listed in RG 1.147 as alternatives to engineering standards for the construction, inservice inspection, and inservice testing of nuclear power plant components. Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1," is listed in RG 1.147, Table 2, "Conditionally Acceptable Section XI Code Cases." The required RG 1.147 Condition associated with Code Case N-702 (Reference 4) is as follows:

> The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of [Nuclear Regulatory Commission] NRC Safety Evaluation [SE] regarding [Boiling Water Reactor (BWR) Vessel and Internals Project] BWRVIP-108 dated December 18, 2007 (ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 (ML13071A240) are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.

In the section of the FRN associated with *NRC Responses to Public Comments on Draft Regulatory Guides,* the NRC responses to comments specific to Code Case N-702 start on page 9 of 40 (79 FR 65783). An excerpt from the FRN is included as follows:

> Licensees who plan to request relief from the ASME Code, Section XI requirements for RPV nozzle-to-vessel shell welds and nozzle inner radius sections may reference the BWRVIP-241 report as the technical basis for the use of ASME Code Case N-702 as an alternative. However, licensees should demonstrate the plant-specific applicability of the BWRVIP-241 report to their units in the relief request by addressing the conditions and limitations specified in Section 5.0 of the NRC Safety Evaluation for BWRVI P-241.

The proposed alternative provides an acceptable level of quality and safety based on the technical content of BWRVIP-108 and BWRVIP-241, as endorsed by the NRC SEs.

5. Proposed Alternative and Basis for Use

Pursuant to 10 CFR 50.55a(z)(1), relief is requested from performing the required examinations on 100% of the identified nozzle assemblies in Table 1 above. As an alternative, for all welds and inner radii identified in Table 1, Energy Northwest proposes to examine a minimum of 25% of the nozzle-to-vessel welds and inner radius sections, including at least one nozzle/inner radius section from each system and nominal pipe size, in accordance with Code Case N-702 (Reference 4). For the components identified in Table 1, this would mean at least one nozzle/inner radius section from each of the groups identified in Table 2 will be examined.

| **Table 2** | | |
|---|---|---|
| **Group** | **Total Number** | **Number to be Examined** |
| RRC Outlet (N1) | 2 | 1 |
| RRC Inlet (N2) | 10 | 3 |
| Main Steam (N3) | 4 | 1 |
| Core Spray (N5, N16) | 2 | 1 |
| Reactor Low Pressure Injection (LPCI) (N6) | 3 | 1 |
| Top Head Nozzles (N7, N8, N18) | 3 | 1 |
| Jet Pump (N9) | 2 | 1 |

Code Case N-702 stipulates that VT-1 examination may be used in lieu of the volumetric examination for the inner radii (Item No. B3.100). Energy Northwest will utilize Code Case N-648-1 with associated required RG 1.147 Conditions if VT-1 examinations are performed in lieu of volumetric examinations.

The Basis for Use is as follows:

Electrical Power Research Institute (EPRI)Topical report BWRVIP-241, “BWR Vessel and Internals Project Probabilistic Fracture Mechanics Evaluation for the Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii” (hereafter referred to as BWRVIP-241) (Reference 2), documents supplemental analyses for BWR 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, safety evaluation (SE) (Reference 5) for the BWRVIP-108NP report, “BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii”. The BWRVIP-108NP (Reference 1) report contains the technical basis supporting American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) Case N-702, “Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds,” 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.

EPRI Report BWRVIP-241 received a final NRC SE on April 19, 2013 (ML13071A240) (Reference 6). In the SE, Section 5.0 "Conditions and Limitations" indicates that each licensee who plans to request relief from the ASME Code, Section XI requirements for

RPV nozzle-to-vessel shell welds and nozzle inner radius sections may reference BWRVIP-241 report as the technical basis for the use of ASME Code Case N-702 as an alternative. However, each licensee should demonstrate the plant-specific applicability of the BWRVIP-241 report to their units in the relief request by demonstrating all of the following:

(1) The maximum Reactor Pressure Vessel (RPV) heatup/cooldown rate is limited to less than 115°F per hour.

The Columbia Technical Specification limits the heatup/cooldown rate to less than or equal to 100 °F in any one hour period.

For the Recirculation Inlet Nozzles (N2) the following criteria must be met:

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

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

For the Recirculation Outlet Nozzles (N1) the following criteria must be met:

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

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

The terms to be used in the NRC SE Section 5 applicability evaluations criteria 2-4 are:

$C_{RPV}$ = recirculation inlet nozzles N2 (from BWRVIP-241 model) = 19332
$C_{NOZZLE}$ = recirculation inlet nozzles N2 (from BWRVIP-241 model) = 1637
$C_{RPV}$ = recirculation outlet nozzles N1 (from BWRVIP-241 model) = 16171
$C_{NOZZLE}$ = recirculation outlet nozzles N1 (from BWRVIP-241 model) = 1977
p = RPV normal operating pressure (psi)
r = RPV inner radius (inch)
t = RPV wall thickness (inch)
$r_i$ = Nozzle inner radius (inch)
$r_o$ = Nozzle outer radius (inch)

Table 3 below summarizes these results.

| Table 3 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Outlet Nozzles (N1) | | | | | | | | |
| $C_{nozzle}$ | $C_{RPV}$ | p | r | $t_{(min)}$ | $r_i$ | $r_o$ | Criteria (4) ≤1.15 | Criteria (5) ≤1.59 |
| 1977 | 16171 | 1035 | 127 | 9.5 | 10.8 | 15.4 | 0.86 | 1.54 |
| | | | | | | | | |
| Inlet Nozzles (N2) | | | | | | | | |
| $C_{nozzle}$ | $C_{RPV}$ | p | r | $t_{(min)}$ | $r_i$ | $r_o$ | Criteria (2) ≤1.15 | Criteria (3) ≤1.47 |
| 1637 | 19332 | 1035 | 127 | 9.5 | 5.8 | 10.0 | 0.72 | 1.27 |

The results in Table 3 show that Columbia meets the conditional requirements established in Section 5.0 of the NRC SE. (Reference 6) Additionally, Columbia evaluated operational experience (OE) from Electrical Power Research Institute (EPRI) to the BWRVIP Committee Members dated August 31, 2012 (Reference 9) regarding fluence assumptions in BWRVIP-108NP (Reference 1) and determined it was applicable to Columbia's N6 nozzle. In response Columbia had a plant specific analysis performed (Reference 10) to verify that neither the thermal cycles nor the fluence level would adversely impact the outcome of the probabilistic fracture mechanics analysis that formed the basis of BWRVIP-108NP. The plant specific analysis shows that the CGS Plant N1 nozzles meet the acceptable failure probability even when considering fluence levels predicted in the beltline region to 60 years of operation, this bounding analysis qualifies all RPV nozzles with full penetration welds (except feedwater and control rod drive return nozzles) for reduced inspection using ASME Code Case N-702 to the end of the period of extended operation.

A review of the most recent examination results for the components listed in Table 1 show no recordable indications or indications exceeding ASME limits have been detected. Greater than 90% examination coverage has been achieved on all of the examinations with the exception of N3-72, N3-252, N3-288, N5-120, N6-45, N6-135 and N18 which had limited coverage relief granted by the NRC. (See References 13 and 14 respectively.)

Based upon the above information, the RRC inlet and outlet nozzles meet the NRC SE criteria as set forth in Reference 6 and therefore Code Case N-702 is applicable. The RPV is low alloy steel plate specification SA-533 grade B class I, the nozzles are low alloy steel forging specification SA-508 class 2 and the weld metal used in the welds specified in Table 1 is carbon/low alloy steel which are the typical materials identified in BWRVIP-108NP therefore, the BWRVIP-108NP evaluation is applicable and appropriate. A bounding plant specific analysis shows that the nozzles meet the acceptable failure probability even when considering fluence levels predicted in the beltline region to 60 years of operation. (Reference 10) A review of the inspection history shows no unacceptable indications reported to date. Therefore, use of Code Case N-702 provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(z)(1) for all Inlet and Outlet nozzle-to-vessel shell welds and nozzle inner radii sections identified in Table 1.

6. Duration of Proposed Alternative

The duration of this request is for the fourth ten-year inservice inspection interval ending December 12, 2025.

7. Precedents

There are two precedents for this request in the NRC SEs approving Columbia's third 10-year ISI interval relief requests 3ISI-09 and 3ISI-14 transmitted by References 7 and 8, respectively. This relief request for the fourth interval combines the two approved relief requests from the third interval.

Similar relief requests were granted to LaSalle County Station (RR I3R14) Units 1 and 2 and Cooper Nuclear Station (RI-08) (References 11 and 12, respectively).

8. References

1. EPRI, Palo Alto, CA, "BWRVIP-108NP: BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," 1016123, November 2007.

2. EPRI, Palo Alto, CA, "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," 1021005, October 2010.

3. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plants," 2007 Edition through 2008 Addenda.

4. ASME Boiler and Pressure Vessel Code, Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1," February 20, 2004.

5. Matthew A. Mitchell, Office of Nuclear Reactor Regulation, to Rick Libra, BWRVIP Chairman, "Safety Evaluation of Proprietary EPRI Report, 'BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)'," December 19, 2007.

6. Sher Bahadur, Office of Nuclear Reactor Regulation, to Dennis Madison, BWRVIP Chairman, "Final Safety Evaluations of the Boiling Water Reactor Vessel Internals Project (BWRVIP)-241 Report, 'Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend Radii (TAC NO. ME6328)" April 19, 2013.

7. Michael T. Markley, Office of Nuclear Reactor Regulation, to J. V. Parrish, Chief Executive Officer, Energy Northwest, "Columbia Generating Station – Request for Relief No. 3ISI-09 for the Third 10-Year Inservice Inspection Program Interval (TAC NO. MD9850)," dated April 8, 2009."

8. Eric R. Oesterle, Office of Nuclear Reactor Regulation, to Mark E. Reddemann, Chief Executive Officer, Energy Northwest, "Columbia Generating Station – Request for Alternative 3ISI-14 to the Requirements of the ASME Code (TAC NO. MF 3435)," dated February 13, 2015.

9. Chuck Wirtz, FirstEnergy, BWRVIP Integration Chairman and Randy Stark, EPRI, BWRVIP Program Manager, to All BWRVIP Committee Members, "BWRVIP Support of ASME Code Case N-702 Inservice Inspection Relief," August 31, 2012.

10. Structural Integrity Associates, Inc. Calculation, "Code Case N-702 Evaluation of the Columbia Generating Station," October 30, 2014 (Columbia CVI/CAL 1012-00,18).

11. Travis L Tate, Office of Nuclear Reactor Regulation, to Bryan C. Hanson, Senior Vice President, Exelon Generation Company, LLC, President and Chief Nuclear Office (CNO), Exelon Nuclear, "LaSalle County Station, Units 1 and 2, Relief from the Requirements of the ASME Code Re: RR I3R14, Proposed Alternative to the Examination Requirements for Nozzle-to-Vessel Welds and Inner Radii Sections in Accordance with 10 CFR 50.55a(z)(1) (TAC Nos. MF5654 and MF5655)," October 30, 2015.

12. Michael T Markley, Office of Nuclear Reactor Regulation, to Oscar A Limpias, Vice President-Nuclear and CNO, Nebraska Public Power District, "Cooper Nuclear Station – Relief Request No. RI-08, Revision 0 Applicable to Fourth 10-Year Inservice Inspection Interval (TAC No. MF4429)," May 20, 2015.

13. Thomas G. Holtz, Office of Nuclear Reactor Regulation, to J. V. Parrish, Chief Executive Officer, Energy Northwest, "Columbia Generating Station - Request for Relief No. 2ISI-32 for the Second 10-Year Inservice Inspection Program Interval (TAC No. MD3905)," December 18, 2007.

14. William H Bateman, Office of Nuclear Reactor Regulation, to J. V. Parrish, Vice President Nuclear Operations, Washington Public Power Supply System, "Evaluation of the Second Ten-Year Interval Inservice Inspection Program Plan and Associated Relief Requests for the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2) (TAC No. M91352)," December 12, 1995.