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Proprietary – Withhold under 10 CFR 2.390. Enclosure 2 contains PROPRIETARY information.

10 CFR 50.55a

April 22, 2020
G02-20-048

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-09**

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(z)(1), Energy Northwest hereby requests the U.S. Nuclear Regulatory Commission's approval of the enclosed relief request beginning with the fourth ten-year inservice inspection program at Columbia Generating Station. The details of the 10 CFR 50.55a request are included in the Enclosure 1.

Enclosure 2 to this submittal contains 1801567.301P, "Probabilistic Fracture Mechanics Evaluation for Columbia Feedwater Nozzle". Structural Integrity Associates, Incorporated considers certain information contained in Enclosure 2 to be proprietary and, therefore, requests that it be withheld from public disclosure in accordance with 10 CFR 2.390. A non-proprietary version of this document is provided in Enclosure 3. Enclosure 4 contains the associated affidavit for the request to be withheld from public disclosure.

When Enclosure 2 is removed from this letter, the letter and remaining Enclosures are NON-PROPRIETARY.

Approval of the relief request is requested within one year of the date of this submittal. Once approved, the relief request shall be implemented within 60 days.

There are no new commitments made in this submittal. If you have any questions or require additional information, please contact Mr. R. M. Garcia, Licensing Supervisor, at 509-377-8463.

Executed this 22 day of April, 2020

Respectfully,

DocuSigned by:
J. Kent Dittmer
60D4E7CA44C64D4...

J. Kent Dittmer
Vice President Engineering

Enclosures: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda - BPA - 1399
EFSECutc.wa.gov – EFSEC
E Fordham – WDOH
R Brice – WDOH
L Albin – WDOH

10 CFR 50.55a Request Number 4ISI-09
Alternative Requirements for Reactor Feedwater Nozzle Inner Radius and Nozzle-to-Shell Weld Examinations

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

Alternate Provides Acceptable Level of Quality and Safety

1. ASME CODE COMPONENT(S) AFFECTED

Component:	Reactor Pressure Vessel (RPV) Reactor Feedwater (RFW) Nozzles
Code Class:	1
Examination Category:	B-D
Item Number:	B3.90 and B3.100
Component Numbers:	The components in Table 1 are affected by this request.

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

2. APPLICABLE CODE EDITION AND ADDENDA

The applicable American Society of Mechanical Engineers (ASME) Section XI Code Edition and Addenda for Columbia Generating Station's (Columbia) fourth ten-year Inservice Inspection (ISI) interval is the 2007 Edition through the 2008 Addenda. Additionally, for ultrasonic examinations, ASME 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 ASME Section XI, Division 1, Subsection IWB, Table IWB-2500-1, "Examination Category B-D, Full Penetration Welded Nozzles in Vessels." Class 1 nozzle-to-vessel welds and nozzle inner radii examination requirements are delineated in Item Numbers 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 proposed alternative provides an acceptable level of quality and safety based on a plant-specific evaluation using a probability fracture mechanics analysis methodology endorsed by the NRC in BWRVIP-108-A, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii," [Reference 1] and BWRVIP-241-A, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," [Reference 2].

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 RFW 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 a minimum of 25% of the inner radii using volumetric inspection methods performed in accordance with ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," as modified by 10 CFR 50.55a.

The Basis for Use is as follows:

Electrical Power Research Institute (EPRI) Technical Reports BWRVIP-108-A [Reference 1] and BWRVIP-241-A [Reference 2] contain the technical basis supporting ASME Boiler and Pressure Vessel Code Case N-702, "Alternative Requirements for Boiling Water Reactor Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1" [Reference 4] for reducing the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radius regions from 100% to 25% of the nozzles for each nozzle type during each ten-year ISI interval. However, the reports and Code Case N-702 explicitly exclude reactor feedwater nozzles stating that these nozzles are managed under a separate mandated program directed by NUREG-0619. This request proposes that the RFW nozzle to shell welds and inner radii examinations mandated under NUREG-0619 can be subsumed by the current ASME Section XI requirements and the number of inspections reduced based on this relief request.

Discussion on NUREG-0619

Columbia complies with the inspection requirements of NUREG-0619 by implementation of the NRC approved alternative GE-NE-523-A71-0594-A, [Reference 3], which stipulates inspections for the RFW nozzle inner radii, nozzle inner bore regions, and the spargers. The nozzle inner radii inspections are done in compliance with ASME Section XI, as modified by the PDI Program description and meets the established criteria of Reference 3. Section 6.3 of Reference 3 states that after compliance with ASME Section XI, Appendix VIII, the examination frequency will be the ASME Section XI examination frequency for non-interference fit plants such as Columbia.

NUREG-2221 "Technical Bases for Changes in the Subsequent License Renewal Documents NUREG-2191 and NUREG-2192," December 2017, states that the recommendation for condition monitoring of the RFW nozzles be performed under ASME Section XI Inservice Inspections for the condition monitoring basis for managing cracking in Boiling Water Reactor (BWR) feedwater nozzles induced by cyclical loading mechanism. This is based in part on improvements in ASME Section XI directed volumetric examination techniques which now meet the requirements proposed in Reference 3. Thus there is a general acknowledgement that the RFW nozzle examinations need no longer go beyond ASME Section XI guidelines and a separate mandated program is not required.

Justification for Reduction to 25 Percent

Using the same analytical methodology as employed in BWRVIP-108-A and BWRVIP-241-A [References 1 and 2] it can be shown that Columbia's RFW nozzle failure probabilities due to a low temperature over pressure (LTOP) event at the nozzle radius region and the nozzle-to-vessel shell weld are very low and meet the Nuclear Regulatory Commission (NRC) acceptance criteria in NUREG-1806. Based 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 plant-specific probabilistic fracture mechanics (PFM) analysis [Reference 5] employed a Monte Carlo simulation using Structural Integrity Associates, Inc. proprietary software VIPERNOZ, which was developed for RPV nozzle weld inspections with BWRVIP-108-A. Columbia's nozzle stresses are used with probabilistic distributions from BWRVIP-108-A and BWRVIP-241-A [References 1 and 2] to evaluate the plant specific probabilities. Proprietary and Non-Proprietary versions of the analysis are provided in Enclosures 2 and 3 of this request.

The conclusion of the analysis [Reference 5] is:

For normal operation for all paths at the nozzle blend radius and the nozzle-to-shell weld, no failures occurred in 1 million simulations for 60 years with 25% inspection. The probability of failure (PoF) for normal operation is calculated to be less than 1 failure / 1 million simulations / 60 years = 1.67×10^{-8} per year. The calculated PoF for normal operation for all paths is less than the allowable PoF of 5×10^{-6} per year and meets the acceptance criterion from NUREG-1806.

For LTOP events for Path 5 at the nozzle-to-shell weld, three (3) LTOP failures occurred in 1 million simulations for 60 years with 25% inspection. The conditional PoF for LTOP events for this path is calculated to be 3 failures / 1 million simulations / 60 years = 5.0×10^{-8} per year. Accounting for an LTOP event occurrence of 1×10^{-3} per year, the calculated PoF for LTOP events for this path is 5.0×10^{-11} per year, 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 paths at the nozzle blend radius and the nozzle-to-shell weld, no failures occurred in 1 million simulations for 60 years with 25% inspection. Thus, for all other paths, the calculated PoF for LTOP events is less than 1.67×10^{-11} per year, which is less than the allowable PoF of 5×10^{-6} per year and meets the acceptance criterion from NUREG-1806.

Thus, the feedwater nozzles are qualified for reduced inspection through the period of extended operation using the same methodology as ASME Code Case N-702.

It is noted that Code Case N-702, is listed in Regulatory Guide (RG) 1.147, Revision 18, Table 2, "Conditionally Acceptable Section XI Code Cases." The required condition associated with Code Case N-702 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 regarding 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.

The condition is intended to verify the applicability of the technical basis documents [References 1 and 2] to the licensee's design. Because the technical basis documents exclude the RFW nozzles the Code Case is not applicable, however, Energy Northwest feels the intent of the Code Case, the condition, and acceptance criteria are demonstrated by the plant-specific analysis for Columbia.

Columbia Operating Experience

A detailed evaluation of the historical problems of the RFW nozzle and sparger is presented in NEDE-21821, "BWR Feedwater Nozzle/ Sparger Final Report," March 1978. 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. Columbia implemented these changes during construction including clad elimination around the nozzle and a welded thermal sleeve and safe end design.

A review of the most recent examination results for each component listed in Table 1 show no recordable indications in the nozzle-to-vessel welds or the inner radii. All of the examinations were volumetric, using ultrasonic testing methodology performed in accordance with ASME Section XI, 2001 Edition (or later) and as modified by the Performance Demonstrated Initiative (PDI) program and 10 CFR Part 50 requirements. All of the examinations had greater than 99% examination coverage.

Industry Operating Experience

NUREG-0619 was published in November 1980 to address instances of thermal fatigue cracking of the RFW and Control Rod Drive Return Line Nozzle. The guidance gave design, operating, and inspection recommendations to address these concerns. As identified in a July 25, 2006 BWRVIP letter to the NRC (Accession Number ML062080159), a survey of all U.S. BWRs shows the majority of RPV nozzles had no reportable indications in the nozzle-to-shell weld or their inner radii. A few nozzles contained subsurface indications which were determined to be acceptable. The data in the letter indicates that the inspections performed as of that date, using reliable techniques, have shown that there are no active degradation mechanisms for the nozzle-to-shell welds and inner radii regions. A search of the Institute of Nuclear Power Operations (INPO) database performed in January of 2020 found that this trend appears to continue with no reports of indications in RPV nozzle-to-shell welds or inner radii regions found.

6. PROPOSED ALTERNATIVE

Columbia proposes to inspect 25% of the RFW nozzle-to-shell welds and inner radius regions each ten-year ISI inspection interval using volumetric inspection methods which meet or exceed the requirements of ASME Section XI, 2007 Edition with 2008 Addenda, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," as modified by 10 CFR 50.55a. The plant-specific evaluation performed for Columbia's RFW nozzles shows that for a 60-year plant life, the failure probabilities due to a LTOP event at the nozzle inner radius region and the nozzle-to-shell weld are very low and meet the NRC safety goals. Hence the alternative provides an acceptable level of quality and safety pursuant to 10 CFR 50.55a(z)(1) for the RFW nozzle-to-vessel shell welds and nozzle inner radii sections identified in Table 1.

7. DURATION OF PROPOSED RELIEF ALTERNATIVE

The duration of this request is for the remainder of plant life including the period of extended operation ending December 12, 2043.

Columbia is aware that the NRC intends to prohibit use of Code Case N-702 for the period of extended operation in its proposed revision 19 to RG 1.147. Columbia notes that the RFW nozzles are excluded from Code Case N-702 and that in the proposed rulemaking for the revision (Ascension Number ML18099A051) the NRC states "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 Requirements of the License Renewal Rule (10 CFR 54.21),' approved on April 26, 2017, or plant-specific probabilistic fracture mechanics analyses." Columbia believes the plant-specific analysis performed based on the BWRVIP-241-A methodology satisfies the concern of the prohibition.

8. ACRONYMS

ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CFR	Code of Federal Regulations
EPRI	Electric Power Research Institute
ISI	Inservice Inspection
INPO	Institute of Nuclear Power Operations
LTOP	Low Temperature over Pressure
NRC	Nuclear Regulatory Commission
PDI	Performance Demonstrated Initiative
PFM	Probabilistic Fracture Mechanics
PoF	Probability of Failure
RFW	Reactor Feedwater
RG	Regulatory Guide
RPV	Reactor Pressure Vessel

9. PRECEDENTS

None.

10. REFERENCES

1. BWRVIP-108-A, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI, Palo Alto, CA 2018, 3002013092.

2. BWRVIP-241-A, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI, Palo Alto, CA: 2018, 3002013093.

3. GE-NE-523-A71-0594-A Revision 1 "Alternate BWR Feedwater Nozzle Inspection Requirements," May 2000

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. Structural Integrity Associates, Inc. Calculation, "Probabilistic Fracture Mechanics for Columbia Feedwater Nozzle", Revision 1, File No.: 1801567.301P

Enclosure 4

Affidavit for Withholding

Structural Integrity Associates, Inc.

March 27, 2020

Attention:
Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the Following Commercial Document:

1) SI Calculation No. 1801567.301P, Revision 1

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the information identified in the enclosed Affidavit consisting of the commercial information owned by Structural Integrity Associates, Inc. ("SIA") identified above (the "Calculations"). Copies of the Calculations and the Affidavit in support of this request are enclosed.

SIA desires to disclose the Calculations in confidence to fulfill commitments outlined in Energy Northwest Contract 355228 (SIA Project 1801567.00). The Calculations are not to be divulged to anyone outside of the NRC nor shall any copies be made of the Calculations provided herein.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (408) 833-7357. Questions on the content of the Calculations should be directed to me as well.

Sincerely,

Kevin Wong

Kevin Wong
Project Manager

AFFIDAVIT

RE:Request for Withholding of the Following Commercial Documents:

1) SI Calculation No. 1801567.301P, Revision 1

I, Kevin Wong, being duly sworn, depose and state as follows:

I am the Project Manager at Structural Integrity Associates, Inc. whose principal office is located at 5215 Hellyer Ave., Suite 210, San Jose, CA 95138 ("SIA") and I have been specifically delegated responsibility for the above-listed Calculations that is sought under this Affidavit to be withheld (the "Calculations"). I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Calculations on behalf of SIA.

SIA requests that the Calculations be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Calculations are owned by SIA and constitutes commercial information which has not been placed in the public domain by SIA.

b. SIA made a substantial economic investment to develop the Calculations and, by prohibiting public disclosure, SIA derives an economic benefit. The Calculations are entitled to the protection of the United States copyright laws. If the Calculations were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry at no cost, these entities would be able to use the Calculations for their own commercial benefit and profit and without expending the substantial economic resources required of SIA to develop the Calculations.

c. SIA made a substantial investment of both money and employee hours over an extended period of time in the development of the Calculations. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Calculations are highly valuable to SIA.

d. A public disclosure of the Calculations would be highly likely to cause substantial harm to SIA's competitive position. If a party does contract with SIA to obtain the information contained herein, it would require an investment of money, time and effort equivalent to that expended by SIA for the party to duplicate the Calculations.