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February 26, 2020
G02-20-021

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-08**

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(z)(1), Energy Northwest hereby requests the US Nuclear Regulatory Commission's approval of the enclosed relief request related to 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.

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 Ms. D. M. Wolfgramm, Licensing Supervisor, at 509-377-4792.

Executed this 26 day of February, 2020

Respectfully,

J. Kent Dittmer
Vice President Engineering

Enclosure: 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-08
Request for Alternative to VT-3 Visual Examination of Accessible Areas of Reactor Vessel Interior

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

Alternate Provides Acceptable Level of Quality and Safety

1. AMERICAN SOCIETY OF MECHNICAL ENGINEERS (ASME) CODE COMPONENT(S) AFFECTED

Code Class: 1

Examination Category: B-N-1 and B-N-2

Item Numbers: B13.10, B13.20, B13.30, and B13.40

Components Affected: See list below

ASME Category	ASME Item No	Component Identification	Description
B-N-1	B13.10	RPV INTERIOR	Reactor Pressure Vessel Interior
B-N-1	B13.10	AHC-000	Access Hole Cover
B-N-1	B13.10	AHC-180	Access Hole Cover
B-N-2	B13.20	JP01/02 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP03/04 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP05/06 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP07/08 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP09/10 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP11/12 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP13/14 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP15/16 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP17/18 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld
B-N-2	B13.20	JP19/20 RB ATTACH WELD	Jet Pump Riser Brace Attachment weld

ASME Category	ASME Item No	Component Identification	Description
B-N-2	B13.20	SPEC HOLDER ATT WELD 030	Specimen holder Bracket Welds
B-N-2	B13.20	SPEC HOLDER A TT WELD 120	Specimen holder Bracket Welds
B-N-2	B13.20	SPEC HOLDER ATT WELD 300	Specimen holder Bracket Welds
B-N-2	B13.30	STM/DRY SUPPORT 000	Steam Dryer Support Bracket Weld
B-N-2	B13.30	STM/DRY SUPPORT 090	Steam Dryer Support Bracket Weld
B-N-2	B13.30	STM/DRY SUPPORT 180	Steam Dryer Support Bracket Weld
B-N-2	B13.30	STM/DRY SUPPORT 270	Steam Dryer Support Bracket Weld
B-N-2	B13.30	LPCS HDR CLMP ATT WLD 1	Low Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	LPCS HORT SEIS ATT WLD	Low Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	LPCS VERT SEIS A TT WLD	Low Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	LPCS HDR CLMP ATT WLD 2	Low Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	HPCS HDR CLMP ATT WLD 1	High Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	HPCS VERT SEIS ATT WLD	High Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	HPCS HORT SEIS A TT WLD	High Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	HPCS HDR CLMP ATT WLD 2	High Pressure Core Spray Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 005	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 055	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 065	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 115	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 125	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 175	Feedwater Bracket Attachment Weld

ASME Category	ASME Item No	Component Identification	Description
B-N-2	B13.30	FW BRKT ATTACH WELD 185	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 235	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 245	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 295	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 305	Feedwater Bracket Attachment Weld
B-N-2	B13.30	FW BRKT ATTACH WELD 355	Feedwater Bracket Attachment Weld
B-N-2	B13.30	GUIDE ROD ATT WELD 000	Guide Rod Bracket Attachment Welds
B-N-2	B13.30	GUIDE ROD ATT WELD 180	Guide Rod Bracket Attachment Welds
B-N-2	B13.30	STM DRY HD BRKT WELD 000	Steam Dryer Hold Down Bracket Welds
B-N-2	B13.30	STM DRY HD BRKT WELD 090	Steam Dryer Hold Down Bracket Welds
B-N-2	B13.30	STM DRY HD BRKT WELD 180	Steam Dryer Hold Down Bracket Welds
B-N-2	B13.30	STM DRY HO BRKT WELD 270	Steam Dryer Hold Down Bracket Welds
B-N-2	B13.40	RPV CORE SUPPORTS	Reactor Pressure Vessel Core Support Structure

2. APPLICABLE CODE EDITION AND ADDENDA

The fourth Inservice Inspection (ISI) interval code of record for Columbia Generating Station (Columbia) is the 2007 Edition with 2008 Addenda of ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

3. APPLICABLE CODE REQUIREMENT

ASME Section XI, Division 1, Table IWB-2500-1, Examination Category B-N-1, Item number B13.10 and Examination Category B-N-2, Item numbers B13.20, B13.30, and B13.40.

4. REASON FOR REQUEST

This request is to implement ASME Section XI, Division 1, Code Case N-885 in lieu of ASME Section XI, Division 1, Table IWB-2500-1, Examination Category B-N-1, Item number B13.10 and Examination Category B-N-2, Item numbers B13.20, B13.30 and B13.40. This alternative eliminates the vessel interior examination requirement and redefines Item B13.10 as an examination of interior welded attachments within the beltline region. Based on the Electric Power Research Institute (EPRI) assessment [1] this alternative provides an acceptable level of quality and safety.

The remaining ASME Section XI, examinations for the vessel interior attachment welds outside the beltline region and the welded core support structure under Item numbers B13.30 and B13.40 are continued under Code Case N-885 under Item numbers B13.20 and B13.30. Since there is no change to the scope or frequency of these examinations, there is no impact to quality or safety regarding these examinations. ASME Code Case N-885 Item number B13.40 is for removable core support items which is not applicable to Columbia's design. The ASME Code Case was Board approved on December 4, 2018.

5. PROPOSED ALTERNATIVE AND BASIS FOR USE

Columbia proposes to follow the guidance of ASME Code Case N-885 for Category B-N locations in lieu of ASME Section XI, 2007 Edition with 2008 Addenda, Table IWB-2500 Category B-N-1 and B-N-2 examinations, for the remainder of the fourth ten-year ISI interval. This will eliminate the VT-3 visual examination of the vessel internal spaces each period while retaining the VT-1 examination requirement for the accessible interior attachment welds within the beltline region, the VT-3 visual examination of the interior attachment welds outside the beltline region, and the VT-3 visual examination of the welded core support structure each ten year ISI interval.

Adoption of ASME Code Case N-885 provides an acceptable level of quality and safety because 1) the B-N-1 examinations are for foreign material which is not required for vessel integrity; 2) examination of the vessel cladding is not required to support vessel integrity; 3) alternative guidance and practices are in place to address foreign material and debris in the reactor vessel and 4) the remaining B-N-2 examinations applicable to Columbia are continued under the Code Case as shown in comparison table at the end of this section. A detailed discussion of each of these reasons is provided below.

1) Examinations for Foreign Material and Debris Are Not Required for Vessel Integrity

As discussed in the EPRI assessment [1], a review of the historical evolution of the Examination Category B-N-1 requirement within ASME Section XI shows that the purpose of this examination is to detect foreign material and debris. This is clear by the fact that the current B-N-1 examinations are specific to the

accessible “spaces above and below the reactor core,” not the inside surface of the vessel or components as clarified by Table IWB-2500-1, Note 1. This position is supported by ASME Interpretation XI-1-95-27, which states that the B-N-1 examination is intended to look for loose or missing parts and debris but the examination does not include the components within the space. Acceptance Criteria IWB-3520.2(b) and (c) are applicable to these foreign material examinations while the remaining acceptance criteria in the paragraph are applicable to the remaining B-N-2 and B-N-3 components. From this, it is clear that the current ASME Section XI, Item B13.10 examination is for foreign material and debris detection and not intended to address the vessel integrity.

2) ***Examination of the Vessel Cladding Is Not Required to Support Vessel Integrity***

Early versions of ASME Section XI contained a requirement for cladding examinations under category B-I-1. The examination was removed from ASME Section XI with the issuance of the Summer 1976 Addenda. ASME Section XI, 2007 Edition with the 2008 Addenda, Examination Category B-N-1 specifies a VT-3 examination of the “accessible spaces above and below the core” and because some plants have conservatively extended the scope of the B-N-1 examination to include a VT-3 visual examination of accessible regions of the cladding, the EPRI assessment [1] also considered the impact of relevant possible cladding degradation mechanisms to show that these “extended” examinations are not necessary to detect corrosion or cracking of the low-alloy steel vessel.

Industry Operating Experience and Analyses

The relevant cladding degradation mechanisms include; general corrosion, localized corrosion, wear, and cracking of the underlying low-alloy steel. These are referred to as vulnerable regions of cladding in subsequent paragraphs and represent areas where corrosion of the underlying low-alloy steel may occur. As stated in EPRI’s evaluation [1] “Operating experience has been favorable with regard to corrosion of low-alloy steel in [boiling water reactors] BWRs. Several BWRs have operated for decades with unclad areas or areas with intentionally removed cladding, with no evidence of discernable corrosion of the low-alloy steel.” The EPRI report goes on to evaluate the cumulative corrosion, flow accelerated corrosion (FAC) and localized corrosion as possible corrosion damage mechanisms for the low-alloy steel reactor vessels. Based on industry studies, the report estimates a corrosion rate of 0.75 mils per year in a BWR reactor vessel environment. At this rate the cumulative amount of material loss corrosion is 0.06 inches after 80 years. This is comparable to Columbia’s corrosion allowance for reactor pressure vessel interior of 1/16 of an inch. The report also notes that the loss of material due to FAC is not a concern as vessels have minor alloying elements that provide substantial benefit against this form of degradation. Lastly, localized corrosion is largely mitigated in BWRs by effective hydrogen water chemistry and adherence to industry water chemistry guidelines.

This evaluation plus operating experience supports the conclusion that corrosion of low-alloy steel is of low concern for BWRs in the event cladding degradation does occur.

Industry operating experience also indicates that cracking of the underlying low-alloy steel vessel material where the cladding is cracked, damaged, or missing is not likely. Observed cracking that propagated through the cladding over long time periods has typically arrested upon reaching the low-alloy steel. The two primary cracking mechanisms evaluated in the EPRI report are fatigue and pressurized thermal shock (PTS). The primary drive for fatigue in BWR RPVs is thermal fatigue (especially in the region of the feedwater nozzles), which has largely been addressed through design and operational changes. Columbia implemented these changes during construction and has had no reports of thermal fatigue cracking in the RPV during its operating life. Furthermore, existing reactor vessel integrity assessments, which model the potential for shallow surface cracks in the cladding to result in brittle fracture, do not explicitly credit periodic visual examinations of the cladding. Such assessments include the Section XI, Appendix G approach to determining allowable pressure-temperature limits, as well as 10 CFR 50.61 and 10 CFR 50.61a screening criteria for PTS [1]. Thus there is low concern for cracking in low-alloy steel RPVs for BWRs or the need for cladding inspections to detect such degradation.

Columbia Operating Experience

Aside from the ASME Section XI, Category B-N-1 examinations, the following examinations are typically performed at Columbia in and on the reactor vessel during refueling outages. These are visual and volumetric examinations that provide opportunity for detecting any adverse conditions at vulnerable regions of cladding. This includes evidence of damaged cladding that may result from impact, wear, or fretting; low-alloy steel corrosion; or cracking penetrating the cladding.

- ASME Section XI examinations performed during the 10-year ISI Interval include Examination Categories B-A (pressure-retaining welds), B-D (full penetration welded nozzles), and B-N-2 (interior attachments) and;

- BWRVIP Examinations include, but are not limited, to RPV Attachment Welds, Core Spray Piping, Jet Pumps, Shroud Support Welds, and Low Pressure Coolant Injection (LPCI) Couplings.

During Columbia's 2017 refueling outage, while setting up for a jet pump examination, the specimen holder at the 300-degree azimuth was reported as failed resulting in movement of the holder which caused some light scratch marks and slight wear on the reactor vessel cladding. The indications were superficial and the holder was removed to preclude any further degradation. Notification of the event was provided to the NRC on August 28, 2017 (Accession Number ML17240A367). In the spring of 2019, during Columbia's 24th refueling outage, a

VT-3 examination of all accessible regions of the RPV cladding was performed. Minor rub marks and scratches likely from inspection equipment were reported along with some stains but there were no recordable indications. Hence, Columbia's RPV cladding is in satisfactory condition with no known defects or active degradation locations.

Additionally, Columbia performs volumetric examinations of the RPV to nozzle welds and nozzle inner radii in accordance with ASME Section XI, Category B-D requirements as modified by Columbia's relief 4ISI-04 [7]. Columbia's Feedwater, High Pressure Core Spray, Low Pressure Core Spray, CRD Return Line, and Main Steam nozzles are unclad as part of its original design and to date there have been no recordable indications reported at the nozzle weld or inner radii of these locations. Similarly, the Recirculation Inlet, Recirculation Outlet, and Jet Pump Sensing Line nozzles are clad and have had no recordable indications reported to date. This further supports the conclusion that corrosion or cracking of the low-alloy steel is of low concern.

3) Alternative Guidance and Practices Are in Place to Address Foreign Material & Debris in the Reactor Vessel

Debris and foreign material are identified as the leading contributors to fuel rod failures [1]. In response to this and other issues, the industry has developed foreign material exclusion (FME) guidance and work practices to help reduce the amount of foreign objects or debris that may be introduced into the reactor coolant system as a result of human error. This includes guidance for FME work practices published by EPRI [2] and the Institute of Nuclear Power Operations (INPO) [3, 6], as well as practices for detection and removal of foreign material and debris for fuel reliability (e.g., EPRI report 3002010740 [4]).

In practice, examinations for foreign material in the reactor vessel are carried out through foreign object search and retrieval (FOSAR) and core verification maintenance activities performed during every refueling outage. Any loose or missing parts and debris located above the reactor core tend to accumulate on top of the fuel, which is observed during the core verification activities performed at the end of refueling outages. Furthermore, foreign objects and debris are most often identified during other examinations, such as fuel receipt inspection and remote inspection of fuel assemblies as they are offloaded and prior to being reloaded. These routine activities adequately address the concern for the effects of foreign materials or debris within the reactor vessel making examinations for debris under B-N-1 redundant.

Columbia performs several activities in accordance with industry guidance during each refueling outage prior to fuel movement (reload), during visual examinations, and during core verification (after reload) that provide the opportunity to detect and remove debris and foreign material that may threaten the fuel. These activities include:

FOSAR Activity

A FOSAR is performed prior to vessel closure. The task requires a pre-closure visual inspection of the system and material, verifying the requirements of the Internal Cleanliness Class B have been maintained and all Foreign Material, plugs/temporary covers have been removed.

Core Verification Activity

After reload, a visual inspection for foreign material for all fuel bundles in the core is performed. The same retrieval tools that are used for FOSAR activity are available for removal of any debris identified.

Vessel Internals Inspections

Extensive examinations of the vessel internals are performed each outage in accordance with boiling water reactor vessel and internals project (BWRVIP) and ASME inspection requirements that provide opportunities to detect foreign material and debris. These include BWRVIP examinations of the Core Spray, LPCI Coupling, Jet Pumps, Access Hole Covers and Shroud Support welds to name a few and ASME Section XI examinations of the core support structure which will continue under this relief per Code Case N-885. Additionally, examinations are performed in areas made accessible due to maintenance activities such as cell disassembly or jet pump mixer removal [5].

Proposed Alternative

Columbia proposes to follow the guidance of ASME Code Case N-885 for Category B-N locations in lieu of ASME Section XI, 2007 Edition with 2008 Addenda, Table IWB-2500, Category B-N-1 and B-N-2 examinations, for the remainder of the fourth ten-year ISI interval. The EPRI study [1] concluded that the VT-3 visual examination for ASME Examination Category B-N-1 performed once per Section XI inspection period is not necessary. It is not needed or required to ensure the integrity of the vessel or cladding and it is redundant to other foreign material examinations performed during an outage. The interior vessel attachment welds and core support structure continue to be examined under the Code Case such that an acceptable level of quality and safety is maintained. Therefore, Energy Northwest requests that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(z)(1).

Code Case N-885 redefines the ASME Item Numbers but, with the exception of the vessel interior, it does not alter the components required to be examined, the examination method, or the examination frequency. The list below compares the current ASME Section XI requirements and Item Numbers to N-885 to show that the change in Item Numbers is merely administrative. To implement this relief, and avoid unnecessary administrative burden, Columbia will retain the current ASME Category

and Item Number information in the ISI Program Plan and database while eliminating the B-N-1 examination.

Component/Location	ASME Category/ Item No	N-885 Category/ item No.	Method	Frequency
Vessel Interior	B-N-1/ B13.10	NA	NA	NA
Interior Attachment Weld within the beltline	B-N-2/ B13.20	B-N/ B13.10	VT-1 Not changed	Same as first interval. Not changed
Interior Attachment Weld beyond the beltline	B-N-2/ B13.30	B-N/ B13.20	VT-3 Not changed	Same as first interval. Not changed
Core Support Structure	B-N-2/ B13.40	B-N/ B13.30	VT-3 Not changed	Same as first interval. Not changed

6. DURATION OF PROPOSED ALTERNATIVE

The proposed Alternative is requested for the remainder of the fourth Inservice Inspection Interval for Columbia, scheduled to end on December 12, 2025.

7. PRECEDENT

None

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
FAC	Flow Accelerated Corrosion
FME	Foreign Material Exclusion
FOSAR	Foreign Object Search and Retrieval
HPCS	High Pressure Core Spray
ISI	Inservice Inspection
INPO	Institute of Nuclear Power Operations
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
NRC	Nuclear Regulatory Commission
PTS	Pressurized Thermal Shock
RPV	Reactor Pressure Vessel

VT-3 Visual examination meeting the requirements of ASME Section XI IWA-2213

9. REFERENCES

1. EPRI Technical Report 3002012966, “Evaluation of Basis for Periodic Visual Examination of Accessible Areas of Reactor Vessel Interior per Examination Category B-N-1 of ASME Section XI, Division 1,” dated April 2018

2. EPRI Technical Report 3002003060, “Foreign Material Exclusion Process and Methods: Supersedes 1016315,” dated November 2014

3. INPO 07-008, Revision 1, “Guidelines for Achieving Excellence in Foreign Material Exclusion (FME),” dated February 2011

4. EPRI Technical Report 3002010740, “Fuel Reliability Program: Foreign Material Handbook for Improvements in Fuel Performance,” dated November 2017

5. EPRI Technical Report 1009947, “BWRVIP-47-A: BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines,” dated June 2004

6. INPO 07-008 Revision 2, “Guidelines for Achieving Excellence in Foreign Material Exclusion (FME),” dated November 2019.

7. Letter, Robert J. Pascarelli (NRC) to Mark E. Reddemann (Energy Northwest), “Columbia Generating Station – Relief Request for Alternative 4ISI-04 Applicable to the Fourth 10-Year Inservice Inspection Program Interval,” dated October 5, 2016 (ADAMS Accession Number ML16263A233)