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Serial: RA-20-0045 10 CFR 50.55a
March 31, 2020

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

Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324
Renewed License Nos. DPR-71 and DPR-62

Subject: Brunswick Steam Electric Plant RA-20-0045, Request for Alternative to Examination Category B-N-1 (VT-3) Visual Examination of Accessible Areas of the Reactor Vessel Interior

Ladies and Gentlemen,

Pursuant to 10 CFR 50.55a(z)(1), Duke Energy Progress, LLC (Duke Energy) is submitting a request for alternative for the Brunswick Steam Electric Plant (BSEP) to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.” This request is proposed on the basis that the alternative provides an acceptable level of quality and safety.

The Relief Request is provided as an Enclosure to this letter. To support the scheduled Spring 2021, Unit 2 Refueling Outage (B2R25), Duke Energy requests approval of this relief request by February 1, 2021.

This letter contains no new regulatory commitments. Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba, Director – Nuclear Fleet Licensing, at 980-373-2062.

Sincerely,

John A. Krakuszeski
Site Vice President
Brunswick Steam Electric Plant

Enclosure: Relief Request RA-20-0045, Request for Alternative to Examination Category B-N-1 (VT-3) Visual Examination of Accessible Areas of the Reactor Vessel Interior

cc: L. Dudes, USNRC Regional Administrator, Region II
G. Smith, USNRC Sr. Resident Inspector - BSEP
A. Hon, USNRC NRR Project Manager – BSEP

bcc: J. Krakuszeski
C. Nolan
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S. Salazar

Enclosure

Request for Alternative to Examination Category B-N-1 (VT3) Visual Examination of Accessible Areas of the Reactor Vessel Interior

BRUNSWICK STEAM ELECTRIC PLANT
10 CFR 50.55a Request Number RA-20-0045
Request for Alternative to Examination Category B-N-1 (VT-3)
Visual Examination of Accessible Areas of the 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: Class 1

Examination Category: B-N-1

Code Item Numbers: B13.10

Components Affected:

ASME Category	ASME Item No	Component Identification	Description
B-N-1	B13.10	Upper RV 0-90 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Upper RV 90-180 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Upper RV 180-270 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Upper RV 270-360 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 15 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 45 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 75 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 105 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 135 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 165 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 195 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 225 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 255 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 285 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 315 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Annulus Region 345 deg	Reactor Pressure Vessel Interior
B-N-1	B13.10	Rx Vessel Head Interior	Reactor Pressure Vessel Interior

2. APPLICABLE CODE EDITION AND ADDENDA

The fifth Inservice Inspection (ISI) interval code of record for Brunswick Steam Electric Plant (BSEP), Duke Energy Progress, 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 No. B13.10, requires a visual examination (VT-3) of the spaces above and below the reactor core that are made accessible for examination by removal of components during normal refueling outages. This examination requirement is once each inspection period.

4. REASON FOR REQUEST

This request is to eliminate the ASME Section XI, Division 1, Table IWB-2500-1, Examination Category B-N-1, Item number B13.10, visual examination (VT-3) of the spaces above and below the reactor core that are made accessible for examination by removal of components during normal refueling outages. The request will apply a provision of 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. Based on publication of ASME Code Case N-885 [7] and Electric Power Research Institute (EPRI) Technical Report 3002012966 [1], this provision that eliminates B-N-1 (VT-3) visual examinations provides an acceptable level of quality and safety.

This request for alternative does not apply to the requirements of ASME Section XI, Division 1, Examination Category B-N-2, Item No. B13.20, B13.30, or B13.40. These examinations will continue to be performed on the current schedule during the fifth ten-year interval of the BSEP ISI Plan.

In response to request [8], NRC has identified that ASME Code Case N-885 will be considered for inclusion in the proposed Revision 20 of Regulatory Guide 1.147.

5. PROPOSED ALTERNATIVE AND BASIS FOR USE

In accordance with 10CFR 50.55a(z)(1), BSEP proposes to eliminate the Examination Category B-N-1, Item No. B13.10, VT-3 visual examination of spaces above and below the reactor core that are made accessible for examination by removal of components during normal refueling outages. BSEP will continue to perform the Examination Category B-N-2, Item No. B13.20, VT-1 visual examination of accessible welds of the reactor vessel interior attachments within the beltline region of the reactor core. Also, BSEP will continue to perform the Examination Category B-N-2, Item No. B13.30 and B13.40, VT-3 visual examination of accessible welds of reactor vessel interior attachments beyond the beltline region of the reactor core and of the accessible surfaces of the core support structure, respectively. For these continued visual examinations (VT-1 and VT-3), BSEP will continue to apply ASME Section XI, Table IWB-2500, Examination Category B-N-2, for the remainder of the fifth ten-year interval of the BSEP ISI Plan.

Adopting this provision of ASME Code Case N-885 to eliminate the Examination Category B-N-1, Item No. B13.10, VT-3 visual examinations provides an acceptable level of quality and safety because (1) this visual examination of the spaces above and below the reactor core is not required for vessel integrity, (2) the examination of the vessel cladding is not required to support vessel integrity, and (3) procedures and practices are in place to address foreign material and debris in the reactor vessel. A detailed discussion of each of these reasons is provided below.

1) *Examinations of Spaces Above and Below the Reactor Core Are Not Required for Vessel Integrity*

As discussed in the EPRI Technical Report 3002012966 [1], review of the historical evolution of Examination Category B-N-1 within ASME Section XI shows that the purpose of this VT-3 visual examination is to detect foreign material and other debris in the “accessible spaces” of the interier areas above and below the reactor core of the reactor vessel. This purpose is clear by the fact that the B-N-1 examinations are specific to “accessible spaces” inside the reactor vessel and excludes visual examination of the interior surface of reactor vessel welded attachments or components; which is further clarified by Table IWB-2500-1, Note 1.

Table IWB-2500-1, Examination Category B-N-1, Item No. B13.10, apply the acceptance standard in subparagraph IWB-3520.2 to VT-3 visual examination of “accessible spaces” of the reactor vessel interior. Specifically, Acceptance Standard IWB-3520.2(c) is applicable to the visual examination of “accessible spaces” of the reactor vessel interior for foreign materials or accumulation of corrosion products. Acceptance Standard IWB-3520(a), (b), (d), (e), and (f) in this subparagraph are not applicable to the examination of “accessible spaces”, but apply to inspection of reactor vessel interior surfaces, welds, and components. To apply the acceptance standard in subparagraph IWB-3520.2, BSEP will continue to perform Examination Category B-N-2 (VT-3) visual examinations of reactor vessel interior surfaces, welds, and components for the remainder of the fifth ten-year interval ISI Plan.

With elimination of Examination Category B-N-1 (VT-3) visual examination of the “accessible spaces” above or below the reactor core, continuing the Examination Category B-N-2, Item No. B13.20, B13.30, and B13.40, (VT-1 and VT-3) visual examinations of reactor vessel interior welded attachments and surfaces of the core support structure provide an acceptable level of quality and safety.

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

The requirement for reactor vessel interior cladding examination (Examination Category B-I-1) was removed from ASME Section XI Code with the issuance of the Summer 1976 Addenda. ASME Section XI, 2007 Edition with 2008 Addenda, Examination Category B-N-1 specifies a VT-3 examination of “accessible spaces above and below the reactor core.” 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 vessel cladding, the EPRI Technical Report 3002012966 [1] also considered the impact of cladding degradation mechanisms to show that the omission of these “extended” visual examinations of the cladding will not have a significant impact on reactor vessel integrity.

Industry Operating Experience and Analyses

The relevant possible cladding degradation mechanisms include general corrosion, localized corrosion, wear, and cracking of the underlying low-alloy steel. These degradation mechanisms are referred to as possible vulnerable regions of vessel cladding and represent areas where corrosion of the underlying low-alloy steel vessel may occur. As stated in the EPRI Technical Report 3002012966 [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 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 due to corrosion is 0.06 inches after 80 years. This is insignificant when compared to the typical vessel wall thickness of BWR reactor pressure vessels (RPVs). The industry study 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.” The EPRI Technical Report 3002012966 [1] and corresponding industry 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 driver 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. Assessments of PTS transients also consider postulated flaws within the cladding or within the reactor vessel base metal (both surface-connected and embedded flaws). The PTS assessments do not explicitly credit periodic visual examinations of the cladding. Thus, there is low concern for cracking in low-alloy steel RPVs for BWRs or the need for cladding inspections to detect such degradation.

Brunswick Nuclear Plant Operating Experience

BSEP performs the following examinations on the components of the reactor vessel during refueling outages. These are visual and volumetric examinations that provide opportunity for detecting any adverse conditions at the vulnerable regions of the vessel cladding. This includes evidence of damaged cladding that may result from impact; wear or fretting; or low-alloy steel corrosion or cracking penetrating the cladding.

- ASME Section XI Inservice Inspections including Examination Category B-A (pressure-retaining welds), Examination Category B-D (full penetration nozzle welds and inner radius sections), and B-N-2 (interior attachments)
- BWRVIP Examinations include, but not limited to, RPV Attachment Welds, Core Spray Piping welds, Jet Pump assembly welds, Shroud Support Welds, Feedwater sparger assemblies, Top Guide Grid Cells and Dry-Tubes.

During Brunswick's refueling outages of the fourth ISI interval (2009 – 2018), VT-3 visual examinations of all accessible regions of the reactor vessel cladding were performed on both units. No recordable indications related to cladding degradation have been found or reported during these examinations.

The EPRI Technical Report 3002012966 [1], corresponding industry operating experience, and Brunswick operating experience support the conclusion that examination of the reactor vessel interior cladding is not required for reactor vessel integrity. With the omission of visual examination of the vessel interior cladding, there is low concern for adverse impact on the reactor vessel integrity.

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. 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 material 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 in EPRI Technical Report 3002003060 [2] and the Institute of Nuclear Power Operations (INPO) Report 07-008 Revisions 1 and 2 [3, 6], as well as practices for detection and removal of foreign material and debris for fuel reliability (e.g., EPRI Technical Report 3002010740 [4]).

Brunswick performs inspection for foreign material in the reactor vessel through foreign material search and retrieval 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 core support structure, which is observed during fuel movement and core verification activities. Other foreign material and debris in the vessel interior and lower head areas are observed during routine inservice inspection activities. Furthermore, foreign material and debris are often identified during the removal of reactor vessel components during the refueling outages. Once foreign material or debris is observed, maintenance practices are established to either remove the foreign material, or evaluate the consequences if not removing, prior to the reactor vessel head closure. These routine inspection and other vessel interior activities adequately address the concern of foreign materials or debris within the reactor vessel, which makes examinations for foreign material and debris under ASME Section XI, Examination Category B-N-1, redundant.

For these reasons, BSEP proposes to apply a provision of ASME Code Case N-885 to eliminate the Examination Category B-N-1 (VT-3) Visual Examination specified in

ASME Section XI, 2007 Edition with 2008 Addenda, Table IWB-2500. BSEP requests authorization to eliminate the Examination Category B-N-1 (VT-3) Visual Examination pursuant to 10 CFR 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

6. DURATION OF PROPOSED ALTERNATIVE

The proposed alternative is requested for the remainder of the Fifth Ten-Year Inservice Inspection Interval for the Brunswick Steam Electric Plant, which is scheduled to end on May 10, 2028.

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
FME	Foreign Material Exclusion
INPO	Institute of Nuclear Power Operations
NRC	Nuclear Regulatory Commission
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. ASME Code Case N-885, Alternative Requirements for Table IWB-2500-1, Examination Category B-N-1, Interior of Reactor Vessel, Category B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessel, Category B-N-3, Removal Core Support Structure, Approved Date: December 4, 2018

8. NRC Response to ASME Request for Including Specific Code Cases in DRAFT Revisions to Regulatory Guides 1.147, 1.84, and 1.193, Letter Dated: January 31, 2020 (ML20013F350)