Beaver Valley Power Station
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10 CFR 50.55a(z)(2)

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

Subject:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Request to Extend Certain Reactor Vessel Inspections From 20 to 22 Years (Request 1-TYP-4-BN-02, Rev. 0)

In accordance with the provisions of 10 CFR 50.55a(z), Vistra Operations Company LLC (Vistra OpCo) hereby requests Nuclear Regulatory Commission (NRC) approval of a proposed alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," subarticle IWA-2430, "Inspection Intervals," and Table IWB-2500-1, "Examination Categories," at Beaver Valley Power Station, Unit No. 1 (BVPS-1).

The proposed alternative would extend the inservice inspection interval from 20 years to 22 years for certain reactor vessel interior attachment welds and core support structure surfaces. A more detailed description of the proposed alternative and supporting information is presented in the attachment.

Vistra OpCo requests approval of the proposed alternative by January 8, 2027.

There are no regulatory commitments contained in this submittal. If there are any questions, or if additional information is required, please contact Jack Hicks, Senior Manager, Licensing, at (254) 897-6725 or jack.hicks@vistracorp.com.

Sincerely,

Robert Kristophel

Attachment:
Beaver Valley Power Station, Unit No. 1, 10 CFR 50.55a Request 1-TYP-4-BN-02, Revision 0

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Beaver Valley Power Station, Unit No. 1
10 CFR 50.55a Request 1-TYP-4-BN-02, Revision 0

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(2)
- Hardship Without A Compensating Increase in Quality and Safety –

1. ASME Code Component(s) Affected

The affected component is the Beaver Valley Power Station, Unit No. 1 (BVPS-1) reactor vessel, specifically, the reactor vessel sub-components listed below. The applicable American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," examination categories and item numbers are also listed. The listed examination categories (hereafter referred to as category) and item numbers are from Subarticle IWB-2500 and Table IWB-2500-1 of ASME BPV Code, Section XI.

Examination Category	Item No.	Description
B-N-2	B13.60	Interior Attachments Beyond Beltline Region
B-N-3	B13.70	Core Support Structure

2. Applicable Code Edition and Addenda

ASME BPV Code, Section XI, 2013 Edition.

3. Applicable Code Requirement

Subsubarticle IWA-2430, "Inspection Intervals," subparagraph (c)(1) states that:

> Each inspection interval may be extended by as much as 1 year and may be reduced without restriction, provided the examinations required for the interval have been completed. Successive intervals shall not extend more than 1 year beyond the original pattern of 10-year intervals and shall not exceed 11 years in length. If an inspection interval is extended, neither the start and end dates nor the inservice inspection program for the successive interval need be revised.

Table IWB-2500-1, "Examination Categories," includes the following reactor pressure vessel visual examinations once each 10-year interval, among other examination requirements:

- Item number B13.60, accessible interior attachment welds beyond the beltline region (category B-N-2)
- Item number B13.70, accessible core support structure surfaces (category B-N-3)

4. Reason for Request

The current BVPS-1 fifth 10-year inservice inspection interval is scheduled to end on August 28, 2028. In accordance with relief request 1-TYP-BN-01 (Accession Number ML 17319A975), the examination category B-N-2 and B-N-3 components are required to be visually inspected prior to August 28, 2028.

However, Vistra Corporation is currently submitting a separate relief request (Relief Request #1-TYP-4-BA-02) that proposes that the reactor vessel pressure-retaining welds and full penetration nozzle welds (Examination Category B-A and B-D) be examined no later than Spring 2030 during the 1R33 refueling outage. The proposed alternative to the ASME BPV Code requirements for the reactor vessel pressure-retaining welds and full penetration nozzle welds (B-A and B-D) is being made pursuant to 10 CFR 50.55a(z)(1) on the basis that the extended inspection interval to end in Spring 2030 during the 1R33 refueling outage provides an acceptable level of quality and safety.

Request 1-TYP-4-BN-02 proposes to extend the fourth inspection interval for visual examination of the interior attachment welds beyond the reactor pressure vessel beltline region (category B-N-2), and visual examination of the accessible core support structure surfaces (category B-N-3) in order to allow deferral of the subject examinations to the same refueling outage (1R33 in Spring 2030) as the category B-A and B-D reactor pressure vessel shell welds and nozzle welds described in request 1-TYP-4-BA-02. The core support structure must be removed to perform both categories of examinations.

Aligning the volumetric examinations of the reactor pressure vessel welds/full penetration reactor pressure vessel nozzle welds (category B-A and B-D reactor pressure vessel welds) with the visual examinations of the interior attachments/core support structure (Category B-N-2 and B-N-3), will result in less core barrel withdrawals, will minimize the risk associated with core barrel lifts, and will provide for efficient scheduling of outage work related to the reactor pressure vessel and internals components

The core support structure must be removed from the reactor vessel to perform the B-N-2 and B-N-3 examinations. The bottom of the core support structure is secured within the reactor vessel by clevis inserts/core support lugs that interface with radial keys on the outside diameter of the core barrel. Because of the close tolerance fit between the clevis inserts/core support lugs and the radial keys on the core barrel, precision lifts are required to remove and replace the core support structure for the B-N-2 and B-N-3 examinations. Every core support structure lift presents risk in that the reactor pressure vessel and reactor pressure vessel internals may be damaged during the lift.

In addition to the risk presented by the precision lifts required to remove and replace the core support structure, movement of the core support structure presents significant radiological risk. Because of the BVPS-1 refueling cavity design, the core support structure rises out of the water when it is moved between the reactor vessel and the storage stand. During movement of the radioactive core support structure, elevated radiation levels have been observed in the BVPS-1 containment building, creating a risk of associated radiation exposure to workers. Removing the core support structure once to perform both the B-N-2/B-N-3 examinations and category B-A and B-D reactor pressure vessel weld examinations in the same outage will maintain radiation exposure as low as reasonably achievable.

Performing category B-N-2 and B-N-3 examinations and the category B-A and B-D examinations during the same refueling outage results in a significant savings in outage duration, radiation exposure, and avoidance of a lift of a heavy close-fit component that has the potential for inflicting damage to itself and reactor vessel surfaces.

5. Proposed Alternative and Basis for Use

Proposed Alternative

As an alternative to ASME BPV Code, Section XI, Subsubarticle IWA-2430 and Table IWB-2500-1, Vistra Corporation proposes to perform the examination of the category B-N-2 reactor vessel interior attachment welds beyond the beltline region and the category B-N-3 reactor vessel core support structure surfaces no later than the 1R33 refueling outage in Spring 2030. Currently, the subject examinations will need to be performed before the end of the 5th ISI interval (August 28, 2028). The proposed alternative inspection would enable the subject examinations to be performed during the1R33 (Spring 2030) refueling outage and would align the subject examinations with the risk-informed extension of the inservice inspection interval for category B-A reactor vessel pressure-retaining welds and category B-D nozzle-to-vessel and nozzle inner radius section welds that is being submitted in relief request 1-TYP-4-BA-02.

If there is an opportunity to perform examinations due to moving the core support structure from the reactor vessel to the permanent storage stand prior to the Spring 2030 refueling outage, the code required VT-3 examinations will be performed on the core support structure and interior reactor vessel attachments in order to fulfill the examination requirements for the extended interval.

BVPS-1 Inspection History and Results

The B-N-2 visual examinations of the reactor pressure vessel interior attachment welds have been performed twice at BVPS-1. The B-N-3 visual examinations of the reactor pressure vessel core support structure have been performed three times at BVPS-1. There were no relevant conditions observed during the B-N-2 and B-N-3 examinations. During the 1986 B-N-3 examination, five indications were observed on the lower internals. The observed indications were surface related, and were identified as scraped areas, minor damage, and minor porosity. The five indications were assessed and found not to affect the structural integrity of the core support structure. The B-N-2 and B-N-3 examinations were last performed during the 2007 maintenance and refueling outage with no indications observed.

A partial B-N-3 examination was performed during the Spring 2018 refueling outage. This B-N-3 exam was limited because at the time of the examination the lower internals structure was seated on the temporary lower internals storage stand (LISS) that holds it over the reactor vessel. However, examination coverage was achieved on the OD of the lower internals structure except for the components that were below the elevation of the lower core support plate (i.e. the bottom mounted instrumentation structure and the secondary core support) since that part of the lower internals structure remains in the reactor vessel when the lower internals is seated on the LISS. The spring 2018 B-N-3 examination also covered the upper internals assembly, which was located on the storage stand during the inspection. During the 2018 partial examination, no recordable indications were identified except a missing ½” cap screw at the 195-degree thermal shield support block location.

A partial B-N-3 examination was performed during the Spring 2024 refueling outage. This B-N-3 exam was also limited because at the time of the examination the lower internals structure was seated on the temporary lower internals storage stand (LISS) that holds it over the reactor vessel. However, examination coverage was achieved on the ID and OD of the lower internals structure except for the components that were below the elevation of the lower core support plate (i.e. the bottom mounted instrumentation structure and the secondary core support) since that part of the lower internals structure remains in the reactor vessel when the lower internals is seated on the LISS. The spring 2024 B-N-3 examination also covered the upper internals assembly, which was located on the storage stand during the inspection. During the 2024 partial B-N-3 examination, recordable indications were identified on 5 thermal shield flexures (fractured) and two degraded thermal shield support block bolts. Additionally, the degradation previously identified on the 195-degree thermal shield 1/2” cap screw was confirmed.

Category B-N-1 visual examinations were performed during the 2016 and 2019 BVPS-1 refueling outages. The B-N-1 examination includes the reactor vessel interior areas that are made accessible for examination by the removal of components during normal refueling outages. There were no relevant conditions observed during the 2016 and 2019 B-N-1 visual examinations.

In addition to the B-N-1, B-N-2, and B-N-3 examinations, the following examinations have been performed on the BVPS-1 reactor vessel internals as part of the reactor vessel internals aging management program:

- EVT-1 examinations of the core barrel upper flange weld, upper girth weld, lower girth weld, lower flange weld, and control rod guide tube lower flange welds were performed during the Spring 2018 refueling outage with no recordable indications identified.
- Volumetric examinations of the baffle bolts and thermal shield support block bolts were performed during the Fall 2022 refueling outage. 1,076 baffle bolts had no indications, and 12 were not testable due to surface geometry. 2 of 12 TSSBB bolts were failed.
- Visual examinations of the baffle former assembly and baffle edge bolts were performed during the Fall 2022 refueling outage. No indications were identified during these examinations.
- Guide card wear measurements of all of the control rod guide tubes were performed during the Spring 2024 refueling outage. The limiting guide card location had 66.85 EFPY remaining until it reached the red zone.
- Volumetric examinations of the thermal shield support block bolts were performed during the Spring 2024 refueling outage. Inspection results were unchanged from the Fall 2022 inspection.
- EVT-1 examination of the core barrel upper flange weld, upper girth weld, and lower girth weld were performed during the Spring 2024 refueling outage with no recordable indications identified.
- VT-3 examination of the thermal shield flexures were performed during the Spring 2024 refueling outage. 5 of 6 thermal shield flexures had recordable (linear) indications.
- Volumetric examination of the thermal shield support block bolts were performed during the Fall 2025 refueling outage. Inspection results were unchanged from the Fall 2022 and Spring 2024 inspections.
- EVT-1 examination of the core barrel upper girth weld was performed during the Fall 2025 refueling outage with no indications identified.

PWSCC Susceptibility Assessment for Core Support Lugs

The BVPS-1 reactor pressure vessel contains four core support lugs that were fabricated from Alloy 600 base material and attached to the reactor pressure vessel wall with alloy 82/182 welds. Because alloy 600/82/182 material is susceptible to primary water stress corrosion cracking (PWSCC), a PWSCC susceptibility assessment was performed for the core support lugs. Based on low normal operating stresses, low normal operating temperature, a favorable operating environment, and the application of a post weld heat treatment, it is very unlikely that PWSCC will occur in the core support lugs and attachment welds during the operating life of BVPS-1. The core support lug PWSCC assessment is documented in the attached Westinghouse letter LTR-SDA-17-018, Revision 1, "Technical Basis for Beaver Valley Unit 1 Core Support Lug Examination Extension." Reasonable assurance of structural integrity of the core support lugs is provided by the fact that there were no relevant conditions observed during the category B-N-1, B-N-2, and B-N-3 visual examinations that have been performed at BVPS-1, there have been no significant findings in the industry associated with the components present in the BVPS-1 reactor vessel core support lug design, and the attached core support lug PWSCC-susceptibility assessment for BVPS-1 identifies that there is low risk of PWSCC initiation in the BVPS-1 core support lugs during the anticipated operating life of the plant. It should be noted that the PWSCC-susceptibility assessment that is documented in LTR-SDA-17-018, Revision 1 was originally developed to justify a re-inspection frequency of 20 years. However, this assessment is still applicable to the current request for a

22-year interval since the predicted PWSCC initiation times identified in LTR-SDA-17-018, Revision 1 are all well beyond the anticipated operating life of the plant.

Summary

This request would extend the duration of the fourth 10-year inspection interval for category B-N-2 and B-N-3, item numbers B13.60 and B13. 70, visual examinations to the 1R33 (Spring 2030) refueling outage. The proposed inspection interval extension is an alternative to subsubarticle IWA-2430 requirements regarding inspection interval length and Table IWB-2500-1 item numbers B13.60 and B13.70 requirements regarding examination frequency. Removing the core support structure for the sole purpose of performing the B-N-2 and B-N-3 examinations would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

6. Duration of Proposed Alternative

The proposed alternative would extend the duration of the fourth 10-year inspection interval for category B-N-2 and B-N-3, item numbers B13.60 and B13.70, visual examinations to the 1R33 refueling outage in Spring 2030.

7. Precedent

A similar request was proposed for South Texas Project Unit 1, that requested that the visual examinations for Category B-N-2 and B-N-3 components be performed consistent with the proposed inspection interval for Category B-A and B-D volumetric examinations. By letter dated November 7, 2018, (ADAMS Accession Number ML 18303A206), the NRC staff authorized use of the alternative.