UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 28, 2018

Mr. Richard D. Bologna, Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 – SAFETY EVALUATION OF PROPOSED ALTERNATIVES 1-TYP-4-BA-01 AND 1-TYP-4-BN-01 REGARDING THE FOURTH 10-YEAR INTERVAL OF THE INSERVICE INSPECTION PROGRAM (EPIDS L-2017-LLR-0131 AND L-2017-LLR-0137)

Dear Mr. Bologna:

By letters dated October 24, 2017 and November 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17297A318 and ML17319A975, respectively), FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted proposed alternatives, 1-TYP-4-BA-01 and 1-TYP-4-BN-01, for Beaver Valley Power Station, Unit No. 1 (BVPS-1). Supplemental information regarding proposed alternative 1-TYP-4-BA-01 was provided in FENOC's letter dated January 5, 2018 (ADAMS Accession No. ML18005A311).

In these letters, FENOC proposed alternatives to certain inservice inspection (ISI) requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for the fourth 10-year ISI interval for BVPS-1, which began on April 1, 2008, and is scheduled to end on August 28, 2018.

Specifically, in proposed alternative 1-TYP-4-BA-01, the licensee requested to extend the fourth 10-year ISI interval, from August 28, 2018, to August 28, 2028, for examination of certain pressure-retaining reactor pressure vessel (RPV) welds and full penetration RPV nozzle welds. The proposed alternative was submitted pursuant to Title 10 of the *Code of Federal Regulation* (10 CFR) 50.55a(z)(1), on the basis that the alternative results in an acceptably small change in risk. The proposed alternative would allow these weld examinations to be performed during the maintenance and refueling outage currently scheduled for 2027.

In proposed alternative 1-TYP-4-BN-01, the licensee requested to extend the fourth 10-year ISI interval, from August 28, 2018, to August 28, 2028, for examination of certain RPV interior attachment welds and core support structure surfaces. The proposed alternative was submitted pursuant to 10 CFR 50.55a(z)(2) on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The proposed alternative would allow deferral of the subject examinations to the same 2027 maintenance and refueling outage as the examinations associated with proposed alternative 1-TYP-4-BA-01 described above.

The U.S. Nuclear Regulatory Commission staff has reviewed the proposed alternatives as documented in the enclosed safety evaluation. Our safety evaluation concludes the following:

(1) With respect to proposed alternative 1-TYP-4-BA-01, extending the interval for examination of the subject components from 10 years to 20 years will result in no appreciable increase in risk. Therefore, the proposed alternative provides an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(z)(1), the proposed alternative is authorized until August 28, 2028.

(2) With respect to proposed alternative 1-TYP-4-BN-01, extending the interval for examination of the subject components from 10 years to 20 years, provides reasonable assurance of structural integrity of the subject components. In addition, complying with the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(z)(2), the proposed alternative is authorized until August 28, 2028.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the proposed alternatives remain in effect, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Jennifer Tobin, at 301-415-2328 or Jennifer.Tobin@nrc.gov.

Sincerely,

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVES 1-TYP-4-BA-01 AND 1-TYP-4-BN-01

REGARDING FOURTH INSERVICE INSPECTION INTERVAL FOR CERTAIN WELDS

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letters dated October 24, 2017 and November 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17297A318 and ML17319A975, respectively), FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted proposed alternatives, 1-TYP-4-BA-01 and 1-TYP-4-BN-01, for Beaver Valley Power Station, Unit No. 1 (BVPS-1). Supplemental information regarding proposed alternative 1-TYP-4-BA-01 was provided in FENOC's letter dated January 5, 2018 (ADAMS Accession No. ML18005A311).

In the proposed alternatives, FENOC proposed to extend the fourth inservice inspection (ISI) interval from 10 to 20 years for examinations of certain components. These examinations are required by Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code. The proposed alternatives are for the fourth ISI interval at BVPS-1, which began on April 1, 2008, and is scheduled to end on August 28, 2018.

In proposed alternative 1-TYP-BA-01, the licensee requested to extend the fourth 10-year ISI interval, from August 28, 2018, to August 28, 2028, for examination of pressure-retaining reactor pressure vessel (RPV) Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds. The proposed alternative was submitted pursuant to Title 10 of the *Code of Federal Regulation* (10 CFR) 50.55a(z)(1), on the basis that the alternative results in an acceptably small change in risk and provides an acceptable level of quality and safety. The proposed alternative would allow these weld examinations to be performed during the maintenance and refueling outage currently scheduled for 2027.

In proposed alternative 1-TYP-4-BN-01, the licensee requested to extend the fourth 10-year ISI interval, from August 28, 2018, to August 28, 2028, for examination of the Category B-N-2 interior attachment welds within the reactor vessel beltline region and the Category B-N-3 reactor vessel core support structure surfaces. The proposed alternative was submitted pursuant to 10 CFR 50.55a(z)(2) on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of

quality and safety. The requested alternative would allow deferral of the subject examinations to the same 2027 maintenance and refueling outage as the examinations associated with proposed alternative 1-TYP-4-BA-described above.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(4), "Inservice inspection standards requirement for operating plants," require ISI of ASME Code Class 1, 2, and 3 components and system pressure tests occur during the first 10-year interval and subsequent 10-year intervals that comply with the requirements in the latest edition and addenda of Section XI of the ASME BPV Code incorporated by reference in 10 CFR 50.55a(a), subject to the limitations and modifications listed in 10 CFR 50.55a(b). For certain ASME Code Class 1, 2, and 3 components, licensees must meet the requirements of the ASME BPV Code and applicable addenda, except where alternatives have been authorized pursuant to 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2).

Section 50.55a(z) of 10 CFR states that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the U.S. Nuclear Regulatory Commission (NRC or the Commission) if the licensee demonstrates (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Section 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events," of 10 CFR contains requirements for calculating the effects of neutron radiation embrittlement of low-alloy steels.

Regulatory Guide (RG) 1.99, Revision Revision2, "Radiation Embrittlement of Reactor Vessel Materials," describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled RPVs.

In RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," it describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights. Revision 3 that was issued in January 2018 does not affect this safety evaluation (SE).

In RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," it describes methods and assumptions acceptable to the staff for determining the RPV neutron fluence.

In October 2011, the NRC approved Westinghouse topical report WCAP-16168-NP-A, Revision 3 (WCAP-A), "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (ADAMS Accession No. ML113060207), which supports a risk-informed assessment of extensions to the ISI intervals for Categories B-A and B-D components. More information on the use of this topical report is provided below.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternatives requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 1-TYP-4-BA-01

3.1.1 Description of Proposed Alternative

The technical basis supporting the extension of the ISI interval from 10 to 20 years for Categories B-A and B-D components, as stated in proposed alternative 1-TYP-4-BA-01, is documented in WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (ADAMS Accession No. ML082820046). Specifically, WCAP-16168-NP-A, Revision 2 performed risk-informed studies based on probabilistic fracture mechanics (PFM) results for three different pressurized-water reactor (PWR) plants (referred to as the pilot plants): Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) for PWR nuclear power plants in the USA to justify the proposed extension of the ISI interval for Categories B-A and B-D components.

The original SE in WCAP-16168-NP-A, Revision 2, published in 2008, was superseded by an SE dated July 26, 2011 (ADAMS Accession No. ML111600303), to address the PWR Owners Group's (PWROG's) request for clarification of the information needed in applications utilizing WCAP-16168-NP-A, Revision 2. The staff concluded in the July 26, 2011, SE that the methodology presented in WCAP-16168-NP-A, Revision 2 is consistent with RG 1.174, Revision 1, and is acceptable for referencing in requests to implement alternatives to ASME Code inspection requirements for PWR plants in accordance with the limitations and conditions in the SE. To demonstrate that the subject plant parameters and inspection history are bounded by the critical parameters identified in Appendix A in WCAP-16168-NP-A, Revision 2, the licensee's application must provide the following plant-specific information:

(1) Licensees must provide the 95th percentile total through-wall cracking frequency ($TWCF_{TOTAL}$) and its supporting material properties at the end of the proposed 20-year ISI interval. The 95th percentile $TWCF_{TOTAL}$ must be calculated using the methodology in NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (ADAMS Accession No. ML070860156), which is frequently referred as "the NRC PTS Risk Study." The RT_{MAX-X} and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT_{30}, must be calculated using the latest revision of RG 1.99 or other NRC-approved methodology.

(2) Licensees must report whether the frequency of the limiting design basis transients during prior plant operation are less than the frequency of the design basis transients identified in the PWROG fatigue analysis as significant contributors to fatigue crack growth.

(3) Licensees must report the results of prior ISI of RPV welds and the proposed schedule for the next 20-year ISI interval. Each licensee shall identify the years in which future inspections will be performed, and the dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in PWROG letter OG-10-238 (ADAMS Accession No. ML11153A033).

(4) Licensees with B&W plants must (a) verify that the fatigue crack growth of 12 heat up/cool-down transients per year that was used in the PWROG fatigue analysis bounds

the fatigue crack growth for all of its design basis transients and (b) identify the design bases transients that contribute to significant fatigue crack growth.

(5) Licensees with RPVs having forgings that are susceptible to underclad cracking and with $RT_{MAX\text{-}FO}$ values exceeding 240 °F must submit a plant-specific evaluation because the analyses performed in the WCAP are not applicable.

(6) Licensees seeking second or additional interval extensions shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

The WCAP-16168-NP-A, Revision 3, which contains this latter SE for WCAP-16168-NP-A, Revision 2, was issued in October 2011 (ADAMS Accession No. ML11306A084) and is referred to as the WCAP-A henceforth in the SE).

3.1.2 Component(s) for which Alternative is Requested

The affected components are the subject plant RPV welds and full penetration nozzle welds. The following examination categories and item numbers from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code, Section XI, are:

Exam Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Section

3.1.3 ASME BPV Code Requirement for which Alternative is Requested

The ASME BPV Code, Section XI, Paragraph IWB-2412, "Inspection Program B," requires volumetric examination of essentially 100 percent of the total number of RPV pressure-retaining welds identified in Table IWB-2500-1, once each 10-year interval.

3.1.4 Proposed Alternative and Basis for Use

In Request 1-TYP-4-BA-01, the licensee proposed to extend the ISI interval for the ASME Code Category B-A and B-D examination items for BVPS-1 from August 28, 2018, to August 28, 2028. Accordingly, FENOC plans to perform the ASME Code required examination of subject items in 2027, consistent with the schedule proposed in the PWROG letter OG-10-238.

The licensee stated that the proposed alternative is based on an acceptably small change in risk by satisfying the risk criteria specified in RG 1.174. Specifically, the methodology used to conduct this analysis is based on the premises defined in WCAP-A. The topical report focuses on risk assessments of materials within the beltline region of the RPV wall. Appendix A of the WCAP-A identifies the parameters to be used to determine if the pilot plant evaluations in the WCAP-A bound a plant-specific application. The parameters include:

- Dominant PTS Transients in the NRC PTS Risk Study
- TWCF

- Frequency and Severity of Design Basis Transients
- Cladding Layers (single/multiple)

The licensee states that BVPS-1 was the Westinghouse pilot plant used in the WCAP-A, and therefore need not confirm the applicability of the parameters in Appendix A of the WCAP-A for the current license term. However, because the current license term for BVPS-1 referred to in the WCAP-A had expired (due to license renewal in November 2009), FENOC performed reevaluation of these parameters. This reevaluation concluded that the dominant PTS transients in the NRC PTS risk study and the cladding layers remain unchanged during the period of extended operation. Hence, no further evaluation of these two parameters is required for BVPS-1.

For the remaining two parameters, the licensee's reevaluation of TWCF indicates that the revised TWCF value is 1.42E-09 events per year, which is bounded by the WCAP-A TWCF value of 1.76E-08 events per year for PWR plants. The licensee's reevaluation of frequency and severity of design basis transients indicated that for BVPS-1 the projected number of heatup and cooldown cycles for 60 years of operation is 175 transients and the heatup and cooldown design occurrences are 139. Both are less than the bounding seven heatup and cooldown cycles per year.

The licensee noted the satisfactory RPV inspection results for BVPS-1 (Table 2 of Request 1-TYP-4-BA-01).

3.1.5 NRC Staff Evaluation

The NRC staff notes that the proposed alternative requires the volumetric inspection be performed in accordance with an ISI frequency satisfying the requirements of the ASME Section XI Code, IWB-2412 and Table IWB-2500-1.

Since the WCAP-A methodology has already been approved by the NRC staff in the WCAP-A SE, this SE focused on the four critical parameters in Table A-1 of WCAP-A, Appendix A and the six plant-specific information items specified in the WCAP-A SE. The NRC staff confirmed that the licensee's evaluation of the four critical parameters in this proposed alternative is satisfactory because the dominant PTS transients defined in the NRC PTS risk study for BVPS-1 (used in the WCAP-A) are valid for the period of extended operation and the layers of BVPS-1 RPV will not change with time. The remaining two critical parameters are among the six plant-specific evaluated items discussed below.

3.1.5.1 Plant-Specific Information Item (1)

Regarding Plant-Specific Information Item (1) on TWCFs, the licensee followed the guidance in Appendix A of WCAP-A. A summary of the input parameters for all RPV materials and the resulting TWCFs for the controlling materials to demonstrate applicability of the WCAP-A analysis to BVPS-1 is provided in Tables 1A and 1B of the proposed alternative. Specifically, Table 1A of Request 1-TYP-4-BA-01 provides chemistry data, unirradiated reference temperature nil ductility (RT_{NDT}), and neutron fluence values for all RPV materials, and Table 1B provides calculated shift and TWCF for controlling RPV materials. The NRC staff compared Table 1A information with that in the license renewal application (LRA) for BVPS-1 and found three discrepancies: (1) Table 1A contains information for additional RPV materials, (2) the unirradiated RT_{NDT} values for plates in Table 1A are significantly lower than the LRA values, and (3) the neutron fluence values in Table 1A are slightly lower than the LRA values. These LRA

values were accepted in NUREG-1929, "Safety Evaluation Report Related to the License Renewal of Beaver Valley Power Station, Units 1 and 2" (ADAMS Accession No. ML093000278). The discrepancies are evaluated below.

Additional RPV materials

Regarding the first discrepancy, the NRC staff confirmed that these new materials, "Reactor Vessel Extended Beltline Materials" were reported in WCAP-18102, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation" (ADAMS Accession No. ML17284A195). The WCAP-18102 was enclosed in a letter dated October 6, 2017, which requested approval of modified PTS reference temperature values and reactor vessel surveillance capsule withdrawal schedule. These additional materials were included in WCAP-18102 because their fluence is now greater than 1×10^{17} n/cm^2. The NRC staff examined the sum of the shift and the unirradiated RT_{NDT} for each additional RPV material and confirmed that it will not become a new controlling material to replace any controlling material listed in Table 1B of the submittal, nor will it affect the sum of the shift and the unirradiated RT_{NDT} for any Table 1B controlling material. Therefore, Table 1B remains valid, with or without consideration of the extended beltline materials. The October 6, 2017, submittal is currently under a separate review and will not affect this determination.

Significantly lower unirradiated RT_{NDT} values for RPV beltline plates

Regarding the second discrepancy, the NRC staff noted that Charpy test data for RPV beltline plates for BVPS-1 was included in a separate relief request, dated October 6, 2017, to support revision of the unirradiated RT_{NDT} values for these plates. These Charpy test data include results at 70°F for two plates (B6607-2 and B7203-2), which were not reported in the July 8, 1992, response to Generic Letter 92-01, "Reactor Vessel Structural Integrity" (NRC Microfilm Address 62405: 317-358). In the January 5, 2018, supplement the licensee explained that this revised information was reported to the NRC in a letter dated July 6, 2015 (ADAMS Accession No. ML15187A260). The new data is based on Materials Certification information; however, it is not the cause for the significantly lower unirradiated RT_{NDT} values for the four RPV beltline plates (B6607-1, B6607-2, B6903-1, and B7203-2). The significantly lower unirradiated RT_{NDT} values are results of a hyperbolic tangent curve fitting instead of a hand-drawn approach in generating test data-based Charpy curves. The data is acceptable because using hyperbolic tangent curve fitting has become a standard industry practice in the past 20 years. Due to this further explanation, the NRC staff accepts the revised unirradiated RT_{NDT} values for the four plates.

Slightly lower neutron fluence

The neutron fluence reported in Table 1A is slightly lower than the LRA values. The LRA values are for 54 effective full power years (EFPYs) while the Table 1A values are for 50 EFPYs. The approximately 3 percent lower fluence in the proposed alternative is very likely due to the slightly lower EFPYs of 50, indicating that the fluence methodology that was accepted in NUREG-1929 remains unchanged in this proposed alternative.

Table 1B shows that the calculated total TWCF is 1.42E-09 event per year. The TWCF value was obtained by the licensee using the WCAP-A methodology with inputs from Table 1A of the submittal. Except for lower shell axial welds, Table 1A used RG 1.99, Revision 2, Position 1.1 (RG tables instead of surveillance data) to calculate ΔT_{30} (shift + unirradiated RT_{NDT}) for 50 EFPYs for all other RPV beltline materials for BVPS-1. Using these Table 1A values, the

NRC staff has verified the licensee's calculated ΔT_{30} values and the resulting TWCF of 1.42E-09 for BVPS-1. The NRC staff determined that this TWCF can support proposed alternative 1-TYP-4-BA-01 because it is an order of magnitude lower than the value of 1.76E-08 for the Westinghouse pilot plant in the WCAP-A. Hence, the NRC staff concludes that the licensee has addressed Plant-Specific Information Item (1) satisfactorily and that the embrittlement of the BVPS-1 RPV is within the envelope used in the Westinghouse pilot plant analysis.

3.1.5.2 Plant-Specific Information Item (2)

The NRC staff reviewed Plant-Specific Information Item (2) regarding the frequency of the limiting design basis transients and determined that more information was needed regarding the projected number of reactor coolant system heatup and cooldown cycles for 60 years of operation. In Table 4.1-10, "Summary of Reactor Coolant System Design Transients" of the licensee's Updated Final Safety Analysis Report, the 139 heatup and cooldown design occurrences is less than the 175 transients proposed in the alternative.

In the supplement to the proposed alternative dated January 5, 2018, the licensee indicated that the 139 heatup and cooldown events are within the envelope (<200) used in the Westinghouse analysis for the pressurizer surge line to hot leg nozzle and the analytical limit for the other reactor coolant system components is 200 events. Since the 139 heatup and cooldown events are bounded by the WCAP-A analysis for 60 years of operation, the NRC staff determined that the licensee has addressed Plant-Specific Information Item (2) satisfactorily and confirmed that, regarding design basis transients, the WCAP-A methodology is applicable to BVPS-1.

3.1.5.3 Plant-Specific Information Item (3)

Lastly, the NRC staff reviewed Plant-Specific Information Item (3) regarding the results of prior ISI of RPV welds and the proposed schedule for the next 20-year ISI interval. Table 2 in the proposed alternative contains additional information pertaining to previous RPV inspections and the schedule for the future. Specifically, Table 2 indicated that three 10-year ISIs have been performed and that there were seven indications identified in the beltline region during the most recent ISI. Per Table IWB-3510-1 of Section XI of the ASME BPV Code, these indications are acceptable.

Regarding the 10 CFR 50.61a requirements on allowable flaws within the inner 1/10th of the RPV wall thickness, the licensee states that none of these indications are within the inner 1/10th or 1 inch of the RPV wall thickness. Because both ASME BPV Code, Section XI and 10 CFR 50.61a requirements regarding detected flaws are met, the NRC staff determined that the licensee has addressed the first part of Plant-Specific Information Item (3) satisfactorily.

The licensee proposed to conduct the next inspection for the BVPS-1 RPV in 2027. Since the RPV inspection will be conducted in the same year as proposed in the PWROG letter, OG-10-238, prior to the proposed ISI extension date of August 28, 2028 for BVPS-1, the NRC staff determined that the licensee has addressed the second part of Plant-Specific Information Item (3) satisfactorily.

The licensee did not address Plant-Specific Information Items (4), (5), and (6). The NRC staff examined the specifics in each of these Plant-Specific Information Items and confirmed that BVPS-1 does not meet the condition to address the item.

In summary, the NRC staff has reviewed the licensee's proposed alternative and supplement and performed independent calculations to verify the input data and output results in Tables 1A and 1B of the submittal. The difference between the licensee's and staff's calculated $TWCF_{95\text{-}TOTAL}$ is insignificant. The NRC staff used this information to confirm that the proposed alternative is based on the WCAP-A methodology and the $TWCF_{95\text{-}TOTAL}$ value in Table 1B of the submittal is bounded by the corresponding pilot plant parameter in the WCAP-A. Consequently, the licensee has demonstrated that the proposed alternative meets the guidance provided by RG 1.174, Revision 1, for risk-informed decisions and, therefore, will provide an acceptable level of quality and safety.

3.2 1-TYP-4-BN-01

3.2.1 Description of Proposed Alternative

In proposed alternative 1-TYP-4-BN-01, the licensee proposed extending the duration of the fourth 10-year inspection interval for visual (i.e., VT-3) examinations of certain RPV interior attachment welds and core support structure surfaces to August 28, 2028. This extension would allow the VT-3 examinations to be done in 2027, at the same time as when the 1-TYP-4-BA-01 examinations are scheduled (see above).

3.2.2 Components for Which Alternative is Requested

The affected components are the interior attachments and core support structures of the subject plant RPV. The following examination categories and item numbers from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code, Section XI, are addressed in this request:

Exam Category	Item Number	Description
B-N-2	B13.60	Interior Attachments Beyond Beltline Region
B-N-3	B13.70	Removable Core Support Structures

3.2.3 ASME BPV Code Requirement for Which Alternative is Requested

Section IWA-2430 of the ASME BPV Code, Section XI, provides the requirements for inservice examination intervals. Section IWA-2430(d)(1) states that "each inspection interval may be reduced or extended by as much as one year," for components inspected under Program B. Table IWB-2500-1 of the ASME BPV Code, Section XI, provides the examination requirements for Category B-N-2 and B-N-3 components. Table IWB-2500 1 requires VT 3 examinations for Item numbers B13.60 and B13.70 be performed each inspection interval.

3.2.4 Proposed Alternative and Basis for Use

The licensee's request for approval of the proposed alternative provides operating experience (OE) and an overview of a stress corrosion cracking assessment to justify that the proposed alternative would not compromise structural integrity or safety. The licensee's letter states that no relevant conditions have been observed at BVPS-1 during B-N-2 and B-N-3 visual examinations and that the locations were last examined in 2007 with no indications observed. The letter also states that industry OE has not revealed any significant finds associated with components present in the BVPS-1 reactor vessel design. Additionally, the category B-N-1 (interior of reactor vessel) visual exams were performed at BVPS-1 during the 2016 refueling outage and no relevant conditions were observed.

The licensee stated that four Alloy 600 lugs are attached to the BVPS-1 RPV with alloy 82/182 welds. The licensee identified 600/82/182 material as susceptible to primary water stress corrosion cracking (PWSCC) and an assessment was performed. The PWSCC assessment determined that it is unlikely that PWSCC will occur in the core support lugs and attachment welds during the operating life of BVPS-1. This determination is based on 1) the normal operating temperatures and stresses experienced by the lugs; 2) negligible radiation effects and nonaggressive environment; and 3) reduction of residual stresses due to post weld heat treatment (PWHT). The assessment also concluded that two lugs would need to fail to result in a complete loss of lateral constraint.

Additionally, the licensee addresses opportunistic examinations of the B-N-2 and B-N-3 locations. The licensee's request states that, "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 2027 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 exam requirements for the extended interval."

3.2.5 Staff Evaluation

The NRC staff has reviewed the information in the proposed alternative. Historically, the RPV welds and the VT-3 inspections of the core support structures have always been performed at the same time. Given approval of proposed alternative 1-TYP-4-BN-01, there is no reason the core barrel and fuel would need to be removed other than the B-N-2 and B-N-3 inspections. The staff notes that every time the core barrel and fuel are removed from the unit, there will be additional radiation exposure to workers in the area. By deferring the examinations to the same refueling outage as the Categories B-A and B-D RPV shell welds and nozzle welds described in 1-TYP-4-BN-01, radiation exposure to the workers would be reduced such that the principles of as low as is reasonably achievable (ALARA) will be met.

The licensee states that there has been no relevant indications found during previous B-N-2 and B-N-3 inspections and; therefore, none would be expected during the next required inspection; B-N-1 inspections will also continue. Based on the above, the NRC staff determined that not authorizing the alternative would result in a decrease in the level of quality and safety. In addition, the absence of relevant indications during previous inspections provides reasonable assurance of structural integrity.

The NRC staff determined that removal of the core barrel and fuel for the sole purpose of performing the ISI B-N-2 and B-N-3 examinations of the core support structures would increase risk and radiation exposure to plant staff and represents a hardship that comes, without a compensating increase in the level of quality and safety. The additional radiation exposure and risk associated with the removal of core barrel would not compensate for an increase in quality and safety. The licensee's proposed alternative will minimize risk and achieve ALARA. Therefore, based on these considerations, the staff concludes that the licensee's request to defer the ISI B-N-2 and B-N-3 exams for the BVPS-1 RPV until 2027 is acceptable.

3.2.6 Hardship Justification

3.2.6.1 Hardship or Unusual Difficulty

As a result of the staff authorized extension of alternative 1-TYP-4-BA-01 (see above for the RPV Category B-A and B-D examinations) the core support structure would need to be removed

prior to August 28, 2018 (i.e., end of the fourth ISI interval) for the sole purpose of performing the B-N- 2 and B-N-3 examinations. The Category B-N-2 and B-N-3 examinations require removal of the core support structure which 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 tight dimensional tolerances, precision lifts are required when removing and replacing the core support structure. Therefore, every time the core support structure is lifted, there is the potential to impart mechanical damage to the RPV and internals. The core support structure is also highly radioactive. The removal of the core support structure exposes the workers to radiation.

The licensee's proposed alternative will reduce the risk of component damage and maintain radiation exposure to the workers' ALARA. Considering the difficulty associated with the removal of the core support structure for a reduced scope of inspection (i.e., only Category B-N-2 and B-N-3 examinations instead of Category B-A, B-D, B-N-2, and B-N-3 examinations), the NRC staff concludes that performing only Category B-N-2 and B-N-3 examinations present an unusual difficulty or hardship.

3.2.6.2 Compensating Increase in the Level of Quality and Safety

The plant-specific and fleet examinations of the Category B-N-2 and B-N-3 components have not resulted in any significant finds. The licensee identified the 600/82/182 material used to fabricate the core support lugs and welds as susceptible to PWSCC. A PWSCC assessment was provided as part of the request for 1-TYP-BN-01. The PWSCC assessment calculated the maximum operating stresses to be 10.15 thousand pounds per square inch (ksi) and 23.71 ksi for the base metal and weld metal, respectively. The PWSCC initiation thresholds were conservatively estimated to be 20 ksi and 34 ksi for the base metal and weld metal, respectively. The PWSCC assessment also notes that a PWHT was performed, which would reduce the magnitude of fabrication induced residual stresses. Considering the combination of operational and residual stresses, the assessment calculated that the time to PWSCC initiation is greater than the licensed operating period of BVPS-1. The low neutron fluence and relatively low operating temperature also makes PWSCC unlikely to affect the lugs and their attachment welds during the extended ISI interval. The PWSCC assessment further states that cracking in the lugs or their attachment welds has not been observed in any operating plant and that a significant portion of the lug would need to be cracked before failure of the lug would occur. The NRC staff reviewed the PWSCC assessment, including the stress values, operating temperature, PWHT effects, and operating environment, and finds that it provides reasonable assurance that PWSCC, if it exists, will not impact the structural integrity of the core support lugs and attachment welds during the proposed extended ISI interval.

Based on plant-specific and fleet OE, the NRC staff finds that there is reasonable assurance that the B-N-2 and B-N-3 examinations will show favorable inspection results in the extended fourth ISI interval. Therefore, the NRC staff concludes that following the ASME BPV Code requirements would not result in a compensating increase in the level of quality and safety.

3.2.6.3 Opportunistic Inspection if the Core Support Structure is Removed

The hardship associated with the Category B-N-2 and B-N-3 examinations is eliminated if the core support structure is removed from the RPV for a reason other than to solely perform the visual examinations. In the request, the licensee stated that BVPS-1 would opportunistically perform the B-N-2 and B-N-3 examinations if the core support structure is removed. This could decrease the inspection interval of the Category B-N-2 and B-N-3 components to less than 20 years.

The NRC staff considers this aspect of the proposed alternative important because the Category B-N-2 and B-N-3 examination interval of 20 years is based on a qualitative analysis considering unusual difficulty, OE, and PWSCC susceptibility. The 20 year interval for the Category B-A and B-D examinations is based on the quantitative risk-informed analysis of WCAP-16168-NP-A, Revision 3, as referenced above.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that increasing the ISI interval for the ultrasonic testing examination of Categories B-A and B-D components from 10 to 20 years (1-TYP-4-BA-01) will result in no appreciable increase in risk. This conclusion is based on the fact that the plant-specific information provided by the licensee is bounded by the data in WCAP-16168-NP-A, Revision 3, and the proposed alternatives meet all of the conditions and limitations described in WCAP-16168-NP-A, Revision 3. Therefore, proposed alternative 1-TYP-4-BA-01 provides an acceptable level of quality and safety, and the alternative is authorized for Categories B-A and B-D components, pursuant to 10 CFR 50.55a(z)(1), until August 28, 2028.

The NRC staff also concludes that based on 1) the risk to plant structures and ALARA concerns during the removal of the core support structure solely for the Category B-N-2 and B-N-3 examinations; 2) plant-specific and fleet OE; 3) PWSCC assessment; and 4) the potential for opportunistic examinations, requiring the licensee to follow the ASME BPV Code requirements would represent a hardship without a compensating increase in the level of quality and safety. Therefore, proposed alternative 1-TYP-4-BN-01 is authorized for the Category B-N-2 and B-N-3 components pursuant to 10 CFR 50.55a(z)(2), until August 28, 2028.

All other requirements of the ASME BPV Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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Date: March 28, 2018

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 – SAFETY EVALUATION OF PROPOSED ALTERNATIVES 1-TYP-4-BA-01 AND 1-TYP-4-BN-01 REGARDING THE FOURTH 10-YEAR INTERVAL OF THE INSERVICE INSPECTION PROGRAM (EPIDS L-2017-LLR-0131 AND L-2017-LLR-0137) DATED MARCH 28, 2018.

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