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

December 27, 2016

Mr. Marty L. Richey, 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 2 - RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (CAC NOS. MF7212 AND MF7217)

Dear Mr. Richey:

By letter dated December 22, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15356A340), as supplemented by letter dated August 5, 2016 (ADAMS Accession No. ML16221A180), FirstEnergy Nuclear Operating Company (FENOC or the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements at the Beaver Valley Power Station, Unit 2 (BVPS-2).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to extend the third 10-year inservice inspection (ISI) interval for BVPS-2 reactor vessel pressure-retaining Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds from August 28, 2018, to August 28, 2028, on the basis that the alternative provides an acceptable level of quality and safety. This will allow these inspections to be performed during the maintenance and refueling outage currently scheduled in 2027.

Additionally, pursuant to 10 CFR 50.55a(z)(2), the licensee requested to extend the third 10-year ISI interval for 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 to August 28, 2028, 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 allows deferral of the subject examinations to the same 2027 maintenance and refueling outage as the Categories B-A and B-D reactor pressure vessel shell welds and nozzle welds described above.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that FENOC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), for BVPS-2 reactor vessel pressure-retaining Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds, and the alternative is authorized until the end of the third 10-year ISI interval, which is August 28, 2028, for the subject components.

The NRC staff also concludes, as set forth in the enclosed safety evaluation, that FENOC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) for 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. Accordingly, the alternative is authorized until the end of the third 10-year ISI interval, which is August 28, 2028, for the subject components.

All other requirements of the ASME Code, Section XI, not specifically included in the request for the proposed alternatives, remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Michael L. Marshall, Jr., at 301-415-2871 or by e-mail at Michael.Marshall@nrc.gov.

Sincerely,

for

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVES NO. 2-TYP-3-BA-01 REGARDING CATEGORIES B-A AND B-D WELDS AND NO. 2-TYP-3-BN-01 REGARDING CATEGORIES B-N-2 AND B-N-3 EXAMINATIONS OF THE RPV INTERIOR ATTACHMENTS AND CORE SUPPORT STRUCTURE

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated December 22, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15356A340), FirstEnergy Nuclear Operating Company (FENOC or the licensee) proposed two alternatives to the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, ISI Program for Beaver Valley Power Station, Unit 2 (BVPS-2). By letter dated August 5, 2016 (ADAMS Accession No. ML16221A180), FENOC provided revised information regarding the item number of the affected component.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), Enclosure 1 of the licensee's December 22, 2015, letter requests to use Alternative 2-TYP-3-BA-01 on the basis that the alternative provides an acceptable level of quality and safety. This alternative requests an extension of the interval from 10 to 20 years for the performance of ASME Code-required Category B-A (volumetric examinations of the reactor pressure vessel (RPV) welds) and Category B-D (the nozzle inside radius and nozzle weld). The licensee states in its December 22, 2015, letter that this request is based on the methodology defined in Westinghouse Commercial Atomic Power (WCAP)-16168-NP-A, Rev. 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (ADAMS Accession No. ML11306A084), and is consistent with the latest industry implementation plan, Pressurized Water Reactor Owners Group (PWROG) Letter OG-10-238, dated July 12, 2010 (ADAMS Accession No. ML11153A033).

Pursuant to 10 CFR 50.55a(z)(2), Enclosure 2 of the licensee's December 22, 2015, letter requests to use Alternative 2-TYP-3-BN-01 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. This alternative requests an extension of the interval from 10 to

20 years for the performance of ASME Code-required Categories B-N-2 and B-N-3 examinations of the RPV interior attachments and core support structure.

For the third 10-year ISI interval at BVPS-2, the Code of record for the inspection of ASME Code Class 1, 2, and 3 components is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI. The third 10-year ISI interval began on August 29, 2008, and is currently scheduled to end on August 28, 2018.

2.0 REGULATORY EVALUATION

2.1 NRC Regulations and Guidance on ISI

In accordance with 10 CFR 50.55a(g)(4), the licensee is required to perform ISI of ASME Code Class 1, 2, and 3 components and system pressure tests 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 Code, incorporated by reference in 10 CFR 50.55a(a), subject to the limitations and modifications listed in 10 CFR 50.55a(b).

The regulation in 10 CFR 50.55a(z) states, in part, that the U.S. Nuclear Regulatory Commission (NRC) may authorize an alternative to the requirements of 10 CFR 50.55a(g). Either of the two requirements must be met for an alternative to be authorized.
Section 50.55a(z)(1) of 10 CFR requires the licensee to demonstrate that the proposed alternative would provide an acceptable level of quality and safety. Section 50.55a(z)(2) of 10 CFR requires the licensee to show that compliance with the ASME Code 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, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" (ADAMS Accession No. ML003740284), contains guidance 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.

RG 1.174, Rev. 2, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis" (ADAMS Accession No. ML100910006), 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.

WCAP-16168-NP-A, Rev. 2

In June 2008, the PWROG issued the NRC-approved topical report WCAP-16168-NP-A, Rev. 2 (ADAMS Accession No. ML082820046), which is in support of a risk-informed assessment of extensions to the ISI intervals for Categories B-A and B-D components. Specifically, WCAP-16168-NP-A, Rev. 2, took data associated with three different pressurized-water reactor (PWR) plants (referred to as the pilot plants), one designed by each of the three main vendors

(Westinghouse, Combustion Engineering, and Babcock and Wilcox (B&W)) for PWR nuclear power plants in the United States, and performed studies on these pilot plants to justify the proposed extension of the ISI interval for Categories B-A and B-D components from 10 to 20 years.

The analyses in WCAP-16168-NP-A, Rev. 2, used probabilistic fracture mechanics (PFM) tools and inputs from the work described in NUREG-1806, "Technical Basis for Rev. of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report" (ADAMS Accession No. ML061580318), and NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (ADAMS Accession No. ML070860156). The PWROG analyses incorporated the effects of fatigue crack growth and ISI. Design-basis transient data was used as input to the fatigue crack growth evaluation. The effects of ISI were modeled consistent with a previously-approved PFM Code in WCAP-14572-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection" (ADAMS Package Accession No. ML012630375). These effects were considered in the PFM evaluations using ORNL/NRC/LTR-04/18, "Electronic Archival of the Results of Pressurized Thermal Shock Analyses for Beaver Valley, Oconee, and Palisades Reactor Pressure Vessels Generated with the 04.1 version of FAVOR" (the fracture analysis of vessels - Oak Ridge National Laboratory computer code (ADAMS Accession No. ML042960391). All other inputs were identical to those used in the PTS risk re-evaluation underlying 10 CFR 50.61a.

From the results of the studies, the PWROG concluded that the ASME Code, Section XI 10-year inspection interval for Categories B-A and B-D components in PWR RPVs can be extended to 20 years. Their conclusion from the results for the pilot plants was considered to apply to any plant designed by the three vendors as long as the critical, plant-specific parameters (defined in Appendix A of WCAP-16168-NP-A, Rev. 2) are bounded by the pilot plants.

NRC Safety Evaluation for WCAP-16168-NP-A, Rev. 2

The original SE in WCAP-16168-NP-A, Rev. 2, which was published in 2008, was superseded by the July 26, 2011, SE (ADAMS Accession No. ML111600303), to address the December 1, 2009 (ADAMS Accession No. ML093370133), PWROG request for clarification of the information needed in applications utilizing WCAP-16168-NP-A, Rev. 2. The staff's conclusion in this latter SE indicates that the methodology presented in WCAP-16168-NP-A, Rev. 2, is consistent with RG 1.174, Rev. 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. In addition to showing that the subject plant parameters and inspection history are bounded by the critical parameters identified in Appendix A in WCAP-16168-NP-A, Rev. 2, the licensee's application must provide the following plant-specific information:

(1) Licensees must demonstrate that the embrittlement of their RPV is within the envelope used in the supporting analyses. Licensees must provide the 95th percentile total through-wall cracking frequency ($TWCF_{TOTAL}$) and its supporting material properties at the end of the period in which the relief is requested to extend the ISI interval from 10 to 20 years. The 95th percentile $TWCF_{TOTAL}$ must be calculated using the methodology in

NUREG-1874 for determining the reference temperature (RT). The RT_{MAX-X}[1] and ΔT_{30}, (the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level) must be calculated using the methodology documented in the latest revision of RG 1.99, or other NRC-approved methodology. The PWROG response to Request for Additional Information (RAI) 3 from Reference 3 (in the July 26, 2011 SE), and Appendix A in the topical report identifies the information that is to be submitted.

(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 that are considered to significantly contribute 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. The 20-year inspection interval is a maximum interval. In its request for an alternative, each licensee shall identify the years in which future inspections will be performed. 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.

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

(5) Licensees with RPVs having forgings that are susceptible to underclad cracking and with RT_{MAX-FO}[2] values exceeding 240 degrees Fahrenheit (°F) must submit a plant-specific evaluation to extend the inspection interval for ASME Code, Section XI, Categories B-A and B-D RPV welds from 10 to a maximum of 20 years, because the analyses performed in WCAP-16168-NP-A, Rev. 2, 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.

In October 2011, WCAP-16168-NP-A, Rev. 3 (ADAMS Accession No. ML11306A084), which contains the latest SE for WCAP-16168-NP-A, Rev. 2, was issued.

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[1] RT_{MAX-X} is a material property that characterizes the reactor vessel's resistance to fracture initiating from flaws in welds, plates, and forgings. The method of determining RT_{MAX-X} is described in Sections (f) and (g) of 10 CFR 50.61a.

[2] RT_{MAX-FO} means the material property that characterizes the reactor vessel's resistance to fracture F initiation from flaws in forgings that are not associated with welds in the forgings. RT_{MAX-FO} value is calculated under the provisions of Sections (f) and (g) of 10 CFR 50.61a.

3.0 TECHNICAL EVALUATION

3.1 Request for Alternative 2-TYP-3-BA-01

3.1.1 Description of Proposed Alternative

In Alternative 2-TYP-3-BA-01, the licensee has proposed extending the current inspection interval from 10 to 20 years for the ASME Code-required Categories B-A and B-D welds. Under the proposed alternative, the current interval would be extended until August 28, 2028. This extension is consistent with the schedule proposed in the latest industry implementation plan, PWROG Letter OG-10-238.

3.1.2 Components for Which Alternative is Requested

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

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 Shell Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Section

3.1.3 Reason for Proposed Alternative

The proposed alternative will reduce radiation exposure to inspectors and the cost of Code-required examinations.

3.1.4 Basis for Proposed Alternative

The licensee stated that the methodology used to demonstrate the acceptability of extending the inspection intervals for Examination Categories B-A and B-D components is contained in WCAP-16168-NP-A, Rev. 3. This methodology uses the estimated TWCF as a measure of the risk of RPV failure and demonstrates that the inspection interval for the affected components can be extended from 10 to 20 years, meeting the change in risk guidelines in RG 1.174. The licensee addressed the plant-specific information discussed in Section 3.4 of the NRC's SE included in WCAP-16168-NP-A, Rev. 3, as follows:

(1) A plant-specific analysis, with identified critical parameters and detailed TWCF calculations, demonstrated that the RPV parameters are bounded by corresponding pilot plant parameters. The total TWCFs were calculated as 1.31E-11 for BVPS-2, less than the value of 1.76E-08 for the Westinghouse pilot plant in WCAP-16168-NP-A, Rev. 3.

(2) The frequencies of the BVPS-2 RPV limiting design-basis transients are bounded by the frequencies identified in the PWROG fatigue analysis.

(3) The results of the previous RPV inspections for the BVPS-2 RPV were provided. The indications found during these inspections are acceptable as dispositioned by Table IWB-3510-1 of Section XI of the ASME BPV Code (zero reportable indications found with last inspections with methodology based on Section XI and Section V, 1989 Edition and 1995 Addenda), which confirm that satisfactory examinations have been performed on the BVPS-2 RPV. Therefore, the licensee stated that the ISI results are acceptable per the requirements of 10 CFR 50.61a.

The licensee has not addressed plant-specific information items (4), (5), and (6) because they do not apply to Alternative 2-TYP-3-BA-01 for BVPS-2. Since the plant-specific information for the BVPS-2 RPV is bounded by the pilot plant application, the licensee concluded that use of this proposed alternative will provide an acceptable level of quality and safety and, therefore, pursuant to 10 CFR 50.55a(z)(1), requests that the NRC authorize the alternative.

3.1.5 Duration of Proposed Alternative

The proposed alternative would extend the duration of the third 10-year ISI interval for BVPS-2 reactor vessel pressure-retaining Examination Category B-A welds and nozzle-to-vessel and nozzle inner radius section Examination Category B-D welds to August 28, 2028.

3.1.6 Precedents

The licensee noted that this same alternative has been granted by the NRC at the Callaway Plant, Unit No. 1 (ADAMS Accession No. ML15035A148), and Diablo Canyon Power Plant, Unit 1 (ADAMS Accession No. ML15168A024).

3.1.7 NRC Staff Technical Evaluation

Tables 1, 4, and 5 of Enclosure 1 to the licensee's December 22, 2015, letter provide plant-specific information required for application of WCAP-16168-NP-A, Rev. 3, to the BVPS-2 RPV. Tables 2 and 3 of the licensee's letter analyze the available embrittlement data for the plate and weld surveillance materials according to the statistical checks in 10 CFR 50.61a. In both cases, the data pass the statistical checks so that the embrittlement trend curve from 10 CFR 50.61a is applicable without modification. The BVPS-2 RPV has a single-layer cladding on the inside, as assumed in the WCAP-16168-NP-A, Rev. 3, analysis. The NRC staff reviewed the information in the above tables and determined that the licensee has addressed the plant-specific information required by WCAP-16168-NP-A, Rev. 3.

The 95th percentile $TWCF_{TOTAL}$ calculation is shown in Table 5 of Enclosure 1 of the licensee's December 22, 2015, letter. The inputs (fluence, chemistry, and unirradiated properties) are detailed in the table for BVPS-2 at 54 effective full power years, which, according to Table 5.2-10 of the licensee's December 9, 2013 (ADAMS Accession No. ML13344A983), letter, is at the end of plant life extension. The NRC staff compared the inputs to the information provided by the licensee's pressure/temperature limits curves submitted on December 9, 2013, and approved by the NRC on October 15, 2014 (ADAMS Accession No. ML14251A558), and determined that the information is the same. In Table 5, the licensee states that the

methodology in 10 CFR 50.61a was used to calculate ΔT_{30}. Note 6 to Table 5 shows that the 95th percentile $TWCF_{TOTAL}$ was calculated using the methodology of NUREG-1874. Table 1 of Enclosure 1 of the licensee's December 22, 2015, submittal shows that $TWCF_{TOTAL}$ for BVPS-2 RPV (1.31E-11 events per year) is less than the pilot plant (1.76E-08 events per year). Based on the above, the NRC staff determined that the licensee has addressed plant-specific information item (1) satisfactorily. The staff confirmed by review of Table 1 that the embrittlement of the BVPS-2 RPV is bounded by that determined for the Westinghouse pilot plant in WCAP-16168-NP-A, Rev. 3.

To address plant-specific information item (2), the licensee reported the critical parameters for the application of WCAP-16168-NP-A, Rev. 3, for BVPS-2 in Table 1 of the submittal. The NRC staff reviewed the information and notes that the frequency of the limiting design-basis transients during prior plant operation is less than the frequency of the design-basis transients that are considered to significantly contribute to fatigue crack growth, which were identified in the WCAP-16168-NP-A, Rev. 3, fatigue analysis. Based on the above, the NRC staff has determined that the licensee has addressed plant-specific information item (2).

Regarding the indications observed during the previous ISIs (summarized in Table 4), the licensee noted that five indications were found during past inspections within the inner 1/10th, or 1-inch, of the reactor vessel thickness. The indications observed were acceptable per the examination and flaw assessment requirements of 10 CFR 50.61a(e) and Table IWB-3510-1 of Section XI of the ASME Code. The NRC staff finds that the latest ISI UT inspections of the RPV welds at BVPS-2 are typical of what has been observed for other units and are acceptable per the ASME Code. Furthermore, WCAP-16168-NP-A, Rev. 3, does not require the applicant to use Section (e) of the 10 CFR 50.61a rule for the first interval extension; however, for subsequent interval extensions, the applicant will be held to Section (e) of the 10 CFR 50.61a rule.

Based on the above evaluation, the NRC staff concludes that the licensee has addressed plant-specific information item (3) satisfactorily because the licensee demonstrated that the plant-specific flaw information for BVPS-2, Unit 2, in Alternate 2-TYP-3-BA-01 is bounded by WCAP-16168-NP-A, Rev. 3, supporting the plant-specific applicability of WCAP-16168-NP-A, Rev. 3, to BVPS-2.

The proposed alternative inspection would be conducted during a refueling outage in 2027 and will be performed to ASME Code, Section XI, Appendix VIII, requirements. The NRC staff has reviewed the revised PWROG plan contained in PWROG Letter OG-10-238 and finds that the proposed examination schedule is within plus or minus one refueling outage of the date in the PWROG inspection plan for the fleet.

The licensee noted precedents in which the NRC staff authorized this same request at the Callaway Plant, Unit No. 1, and Diablo Canyon Power Plant, Unit 1. The NRC staff determined that these precedents were similar in that the critical parameters for the application of WCAP-16168-NP-A, Rev. 3, as shown in Table 1 of the licensee's December 22, 2015, submittal, were bounded by the pilot plant as they were in the cited precedents.

In summary, the NRC staff has reviewed the licensee's submittal and performed independent calculations to verify the input data and output results in Table 5 of the alternative. The difference between the licensee's and staff's calculated $TWCF_{95\text{-}TOTAL}$ is insignificant. With this

information, the NRC staff concluded that the $TWCF_{95\text{-}TOTAL}$ value in Table 5 of Enclosure 1 is bounded by the WCAP-16168-NP-A, Rev. 3, results. Consequently, the licensee has demonstrated that the proposed alternative will provide an acceptable level of quality and safety and meets the guidance provided by RG 1.174, Rev. 1, for risk-informed decisions.

3.2 Request for Alternative 2-TYP-3-BN-01

3.2.1 Description of Proposed Alternative

In Alternative 2-TYP-3-BN-01, the licensee proposed extending the duration of the third 10-year inspection interval for Categories B-N-2 and B-N-3, Item Numbers B 13.50 and B 13.70, visual examinations to August 28, 2028. This extension would allow the VT-3 examinations to be done in 2027, at the same time as when the Categories B-A and B-D examinations are scheduled.

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 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 Reason for Proposed Alternative

These inspections are typically done at the end of an interval in conjunction with the ultrasonic testing (UT) of the RPV welds, which is normally the only time when there is a full off-load of the core and the core barrel, allowing access to the core support structure. In Alternative 2-TYP-3-BA-01, the licensee has requested to extend the third interval from 10 to 20 years for the Categories B-A and B-D welds, which means that the full off-load of the core and the core barrel will not be done again until 2027. Given the requested extension of the third interval for the Categories B-A and B-D welds, the core support structures may not be accessible for the required VT-3 examinations until 2027.

3.2.4 Proposed Alternative and Basis for Use

Performing the ASME Code-required B-N-2 and B-N-3 inspections requires removal of the core barrel, but to remove the core barrel for the sole purpose of performing these inspections represents an unnecessary risk that would increase the radiation exposure to inspectors if the examinations are not combined with other required maintenance. Therefore, an extension is requested from the Code requirements so that the VT-3 of the core support structures can be performed at the next opportunity when a full core offload and core barrel removal is scheduled at BVPS-2.

The licensee states that the visual examinations of the reactor pressure vessel interior attachment welds (B-N-2) and core support structure surfaces (B-N-3) have been performed twice at BVPS-2, and no relevant indications were noted during the examinations. The licensee

also states that the industry has not found any significant findings relative to the BVPS-2 reactor vessel design. The licensee states that during the 2015 refueling outage, the B-N-1 (including the RPV interior areas made accessible by the removal of components during normal refueling outages) visual examination was performed, and no relevant indications were noted.

This proposed alternative does not change the ASME Category B-N-1 inspections. These visual examinations of the space above and below the core made accessible during normal refueling outages will be performed as required by the ASME Code.

In accordance with 10 CFR 50.55a(z)(2), this extension of the inspection interval from 10 to 20 years for the B-N-2 and B-N-3 inspections is requested on the basis that compliance with the ASME Code-required inspections of the core support structures would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. The fact that no relevant indications were noted during the previous examinations, including the B-N-1 visual examinations during the 2015 refueling outage, provides reasonable assurance of structural integrity.

3.2.5 Duration of Proposed Alternative

The proposed alternative would extend the duration of the third 10-year inspection interval for category B-N-2 and B-N-3, Item Numbers B 13.50 and B 13. 70, visual examinations to August 28, 2028.

3.2.6 Precedents

The licensee noted that this same alternative has been granted by the NRC at the San Onofre Nuclear Generating Station, Unit Nos. 2 and 3 (ADAMS Accession No. ML112730074).

3.2.7 NRC Staff Technical Evaluation

The NRC staff has reviewed the information in Enclosure 2 to the December 22, 2015, submittal. Historically, the RPV welds and the VT-3 inspections of the core support structures have always been performed at the same time. The licensee's request to extend the RPV weld inspection (2-TYP-3-BA-01) was the first part of the December 22, 2015, submittal. Given approval of Alternative 2-TYP-3-BA-01, there is no reason the core barrel and fuel would 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 2-TYP-3-BA-01, radiation exposure to the workers would be reduced such that 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. In addition, B-N-1 inspections will continue. Based on the above, the NRC staff determined that not authorizing the alternative would not result in a compensating increase 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-2 RPV until 2027 is acceptable.

4.0 CONCLUSION

For Alternative 2-TYP-3-BA-01, the NRC staff concludes that increasing the ISI interval for the UT examination of Categories B-A and B-D components from 10 to 20 years 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, Rev. 2, and the requests meet all of the conditions and limitations described in WCAP-16168-NP-A, Rev. 2. Therefore, Alternative 2-TYP-3-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.

Given the approval of Alternative 2-TYP-3-BA-01, the NRC staff concludes that Alternative 1-TYP-3-BN-01 regarding increasing the ISI interval for the VT-3 examination of Categories B-N-2 and B-N-3 components from 10 to 20 years is acceptable, because it minimizes the risk associated with removal of the core barrel and fuel and follows the ALARA principles. Requiring the licensee to follow the ASME Code requirements would represent a hardship that comes, without a compensating increase in the level of quality and safety. In addition, the NRC staff determined that there is reasonable assurance of structural integrity. Therefore, Alternative 2-TYP-3-BN-01 is authorized for Categories 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 Code, Section XI, not specifically included in the request for the proposed alternatives, remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: P Purtscher

Date: December 27, 2016

The NRC staff also concludes, as set forth in the enclosed safety evaluation, that FENOC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) for 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. Accordingly, the alternative is authorized until the end of the third 10-year ISI interval, which is August 28, 2028, for the subject components.

All other requirements of the ASME Code, Section XI, not specifically included in the request for the proposed alternatives, remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Michael L. Marshall, Jr., at 301-415-2871 or by e-mail at Michael.Marshall@nrc.gov.

Sincerely,

/RA REnnis for/

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

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Safety Evaluation

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