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

March 2, 2017

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 NO. MF8854)

Dear Mr. Richey:

By letter dated November 16, 2016 (Agencywide Documents Access and Management System Accession No. ML16323A288), 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 use the proposed alternative to certain requirements associated with repair activities related to the reactor vessel on the basis that the alternative examination provides an acceptable level of quality and safety.

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 under 10 CFR 50.55a and that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that FENOC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative for the third 10-year inservice inspection interval at BVPS-2, or until the replacement of the reactor pressure vessel head, whichever occurs first.

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

If you have any questions, please contact the Project Manager, Taylor Lamb, at 301-415-7128 or Taylor.Lamb@nrc.gov.

Sincerely,

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
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE 2-TYP-3-RV-04, REVISION 1, REGARDING

REACTOR VESSEL HEAD PENETRATION J-GROOVE WELD REPAIRS

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated November 16, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16323A288), FirstEnergy Nuclear Operating Company (FENOC or the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC or the Commission) 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 use the proposed alternative to certain requirements associated with repair activities related to the reactor vessel on the basis that the alternative examination provides an acceptable level of quality and safety. The subject request is a revision to a previously approved request via NRC letter dated June 17, 2016 (ADAMS Accession No. ML16147A362), as corrected by NRC letter dated October 21, 2016 (ADAMS Accession No. ML16228A408), and specifically requests to change the frequency of surface examinations from every outage to every other outage.

2.0 REGULATORY EVALUATION

In this relief request, the licensee proposes to use alternatives to the requirements of paragraph IWA-4000 of the 2001 Edition through 2003 Addenda of Section XI of the ASME Code.

Pursuant to 10 CFR 50.55a(g)(4), throughout the service life of a pressurized water-cooled nuclear power facility, components that are classified as ASME Code Class 1, 2, and 3 must meet the requirements, except the design and access provisions and preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical, within the limitations of design, geometry, and materials of construction of the components. Further, these regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval, and subsequent intervals, comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in paragraph (b) of 10 CFR 50.55a, on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

Alternatives to requirements under 10 CFR 50.55a(g) may be authorized by the NRC pursuant to 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2). In proposing alternatives or requests for relief, the licensee must demonstrate that: (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.

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 proposed alternative requested by the licensee. Accordingly, the NRC staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(z)(1).

3.0 TECHNICAL EVALUATION

3.1 Licensee Relief Request

3.1.1 Component Identification

BVPS-2 Reactor Vessel Head Penetrations 2RCS-REV-21, Numbers 1 through 65

3.1.2 Applicable Edition of the ASME Code

The applicable construction code is the ASME Code, Section III, 1971 Edition through Summer 1972 Addenda.

For the third 10-year inservice inspection (ISI) interval, which began on August 29, 2008, and ends on August 28, 2018, the ASME Code of Record is the 2001 Edition through the 2003 Addenda.

3.1.3 Code Requirements for Which Relief is Requested

ASME Code, Section XI, 2001 Edition through 2003 Addenda, paragraph IWA-4000, contains requirements for the removal of defects from, and welded repairs performed on, ASME Code components. Specific requirements are found for defect removal under subparagraph IWA-4421, which states for the removal or mitigation of defects by welding, subparagraph IWA-4411 must be followed. Subparagraph IWA-4411 requires that repairs and installation of replacement items shall be performed in accordance with the Owner's Design Specification and the original Construction Code of the component or system.

The original Construction Code of the reactor vessel is ASME Code, Section III, 1971 Edition through Summer 1972 Addenda. The licensee requested relief from subparagraphs NB-4131, NB-2538, and NB-2539, which pertain to the removal of base material defects prior to repair by welding, and NB-4451, NB-4452, and NB-4453, which pertain to the removal of weld material defects prior to repair by welding.

3.1.4 Licensee's Proposed Alternative

As an alternative to the defect removal requirements of ASME Code, Section XI and Section III, the licensee proposes the use of the embedded flaw repair process described in Westinghouse

WCAP-15987-P, Revision 2-P-A, "Technical Basis for the Embedded Flaw Process for Repair of Reactor Vessel Head Penetrations," December 2003 (ADAMS Accession No. ML040290246), as detailed in this proposed alternative, for the repair of unacceptable indications in reactor vessel head penetrations and J-groove welds, as approved by the NRC by letter dated July 3, 2003 (ADAMS Accession No. ML031840237). Design and implementation of the repairs will be consistent with WCAP-15987 and WCAP-16158-P, Revision 0, November 2003 (ADAMS Accession No. ML082900208).

Preservice inspections and ISIs of repairs will be consistent with current NRC regulatory requirements rather than those stated in the NRC Safety Evaluation for WCAP-15987 by letter dated April 11, 2003 (ADAMS Accession No. ML030980322), with one additional exception for surface examinations. The NRC safety evaluation specified the use of "Flaw Evaluation Guidelines," which was sent to the Nuclear Energy Institute (NEI) by letter dated April 11, 2003. In lieu of these guidelines, FENOC proposes to follow the current NRC criteria for flaw inspection and evaluation requirements established in 10 CFR 50.55a(g)(6)(ii)(D), which specifies the use of ASME Code Case N-729-1, with conditions, with the one exception that surface examinations would be performed as follows:

> Surface examination of the embedded flaw repair shall be performed to ensure the repair satisfies ASME Code Section III, NB-5350, acceptance standards. The frequency of examination shall be as follows:
>
> (a) Perform surface examination during the first and second refueling outage after installation or repair of the embedded flaw repair.
>
> (b) When the examination results in (a) above verify acceptable results then reinspection of the embedded flaw repair will be continued at a frequency of every other refueling outage. If these examinations identify unacceptable results that require flaw removal, flaw reduction to acceptable dimensions, or welded repair the requirements of (a) above shall be applied during the next refueling outage.

Pursuant to 10 CFR 50.55a(z)(1), the alternative is proposed on the basis that it will provide an acceptable level of quality and safety, while minimizing cumulative occupational radiation exposure (dose).

3.1.5 Licensee's Duration of Relief Request

The proposed alternative will be used for the remainder of the BVPS-2 third 10-year ISI interval, which ends on August 28, 2018, or until the reactor vessel head is replaced, whichever occurs first.

3.1.6 Licensee's Basis for Relief

The licensee states that the purpose of the repair is to embed and isolate identified flaws in the reactor vessel head penetration nozzles and associated J-groove welds. The repair overlay welds are not credited for providing structural strength to the original pressure boundary materials.

The licensee notes that WCAP-15987 describes the embedded flaw repair technique as a permanent repair. The repair is based on the principle that as long as a primary water stress corrosion cracking (PWSCC) flaw remains isolated from the primary water environment, it cannot propagate through the stress corrosion cracking method. Alloy 690 and Alloy 52 are highly resistant to stress corrosion cracking, as demonstrated by multiple laboratory tests, as well as over 18 years of service experience in replacement steam generators. Since Alloy 52 weldment is considered highly resistant to PWSCC, a new PWSCC flaw would not be reasonably expected to initiate and grow through the Alloy 52 repair weld layers to reconnect the primary water environment with the embedded PWSCC flaw.

Additionally, the licensee provided the following clarification:

> In order to provide reasonable assurance that the embedded flaw repairs at BVPS-2 will continue to perform their design function, a combination of volumetric and surface examinations will continue to be performed in accordance with 10 CFR 50.55a and ASME Code Case N-729-1. The volumetric (UT) examination that is performed each outage will continue to monitor the embedded flaw repair for flaw growth or potential leak paths. The surface (dye penetrant) examination will continue to supplement the UT examination when the surface examination is performed every other outage as proposed.

The licensee explains that by the NRC's letter dated June 17, 2016 (ADAMS Accession No. ML16147A362), the NRC staff approved this same repair method for reactor vessel head penetrations and J-groove welds at BVPS-2 for the third 10-year ISI, as described in the current submittal. The only technical difference between the previously approved alternative and the subject request is the frequency of surface examination requirements.

3.2 NRC Staff Evaluation

The NRC previously approved a similar alternative repair method for BVPS-2 by its letter dated June 17, 2016, as modified by letter dated October 21, 2016 (ADAMS Accession No. ML16228A408), for the third 10-year ISI interval. The technical basis, as stated in the previous NRC safety evaluation authorization of the proposed alternative, remains valid to allow BVPS-2 to use the Westinghouse Embedded Flaw Repair Technique (EFR) in lieu of the requirements of IWA-4000 of Section XI of the ASME Code for the third 10-year ISI interval.

The licensee's proposed alternative in this request restates fully the original proposed alternative previously authorized by the NRC, with one additional change. The licensee proposes a change to the previously approved surface examination requirement of each refueling outage to every other refueling outage if two surface examinations performed following installation or repair of the embedded flaw repair have no unacceptable indications. The NRC staff continues to recognize the radiological dose savings of the licensee's proposed alternative surface examination frequency. Further, the NRC staff continues to find that the inspection program in the proposed alternative addresses all known issues with potential fabrication defects associated with the weldability of Alloy 52 before significant degradation of the reactor pressure vessel head or associated nozzles would occur.

The NRC staff reviewed the licensee's request to extend the interval between penetrant test (PT) examinations from every outage to every other outage, if two examinations were performed in which no unacceptable indications were identified. The staff finds this request acceptable based on 1) the lack of PWSCC flaws in EFRs; 2) the excellent service history of Alloy 690/52/152 in other applications to address PWSCC initiation; 3) the continued performance of UT examinations by the licensee during each outage, and 4) the radiological dose involved with performing the PT examinations, which is 0.26 roentgen equivalent man (rem) per nozzle PT examination. Additionally, BVPS-2 is required to perform ultrasonic examinations each refueling outage on each nozzle.

The NRC staff concludes that the performance of PT examinations every outage at BVPS-2 is not necessary to ensure the structural and leaktight integrity of the subject welds if the two previous PT inspections have not found any unacceptable indications. Therefore, the NRC staff finds the licensee's proposed alternative provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative for the third 10-year ISI interval at BVPS-2, or until the replacement of the reactor pressure vessel head, whichever occurs first.

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

Principal Contributor: J. Collins

Date: March 2, 2017

SUBJECT: BEAVER VALLEY POWER STATION, UNIT 2 – RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (CAC NO. MF8854) DATED MARCH 2, 2017

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JCollins, NRR

ADAMS Accession Number: ML17041A185 *by e-mail dated February 2, 2017

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