



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

February 25, 2011

Mr. Paul A. Harden
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. 2 - RELIEF REQUEST
REGARDING AN ALTERNATIVE WELD REPAIR METHOD FOR REACTOR
VESSEL HEAD PENETRATIONS J-GROOVE WELDS (TAC NO. ME4176)

Dear Mr. Harden:

By letter dated June 21, 2010, as supplemented by letter dated August 13, 2010, FirstEnergy Nuclear Operating Company (the licensee) submitted a request for authorization of proposed alternatives to the non-destructive examination acceptance criteria and the filler metal to be used for the part of the repair overlay that extends beyond the J-groove weld and over the stainless steel clad on the inside surface of the reactor vessel head at Beaver Valley Power Station, Unit No. 2 (BVPS-2) for the remainder of the current BVPS-2 10-year inservice inspection (ISI) interval, which ends August 28, 2018. Specifically, the licensee requested to utilize the surface non-destructive examination acceptance criteria of the original construction code versus the previously approved acceptance criteria of no surface indications for the imbedded flaw weld overlay repair technique that extends past the original J-groove weld onto the stainless steel cladding covering the inside surface of the head.

The Nuclear Regulatory Commission (NRC) staff has concluded that compliance with the current requirements would cause an unnecessary burden on the licensee without a compensating increase in the level of quality and safety. Additionally, the NRC staff concluded that the licensee is in compliance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements and the proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to Section 50.55a(a)(3)(ii) of Part 50 of Title 10 of the *Code of Federal Regulations*, the NRC staff authorizes the proposed alternative for the remainder of the current BVPS-2 10-year ISI interval, which ends August 28, 2018.

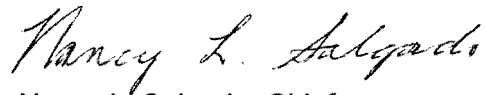
All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

P. Harden

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If you have any questions, please contact the Beaver Valley Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

A handwritten signature in cursive script that reads "Nancy L. Salgado".

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosure:
As stated

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING AN ALTERNATIVE WELD REPAIR METHOD FOR
REACTOR VESSEL HEAD PENETRATIONS J-GROOVE WELDS

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated June 21, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101740436), as supplemented by letter dated August 13, 2010 (ADAMS Accession No. ML102300043), FirstEnergy Nuclear Operating Company (the licensee) submitted a request for authorization of proposed alternatives to the nondestructive examination acceptance criteria and the filler metal to be used for the part of the repair overlay that extends beyond the J-groove weld and over the stainless steel clad on the inside surface of the reactor vessel head at Beaver Valley Power Station, Unit No. 2 (BVPS-2) for the remainder of the current BVPS-2 10-year inservice inspection (ISI) interval, which ends August 28, 2018. Specifically, the licensee requested to utilize the surface non-destructive examination (NDE) acceptance criteria of the original construction code versus the previously approved acceptance criteria of no surface indications for the imbedded flaw weld overlay repair technique that extends past the original J-groove weld onto the stainless steel cladding covering the inside surface of the head.

The U.S. Nuclear Regulatory Commission (NRC) staff previously approved a similar alternative repair method for BVPS-2 by letter dated October 6, 2009 (ADAMS Accession No. ML092700031). During the fall 2009 refueling outage at BVPS-2, the plant's reactor vessel head penetrations and J-groove welds were inspected. Primary water stress-corrosion cracking (PWSCC) indications were identified on two penetrations which required repair. Reactor vessel head penetration number 57's repair did not meet the applicable NDE acceptance criteria of the alternative repair method approved by the October 6, 2009, letter. After several attempts to

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meet the acceptance criteria, the licensee, by letter dated November 14, 2009 (ADAMS Accession No. ML093220057) requested expedited relief from the acceptance criteria specifications of the October 6, 2009, letter. The NRC staff granted verbal relief only for penetration nozzle number 57 on November 15, 2009, documented by a letter dated March 12, 2010 (ADAMS Accession No. ML100680781). At the time of the relief request, a total of approximately 50 rem of personnel radiation exposure was absorbed while conducting the required reactor vessel head inspections and performing the repairs to meet the full NDE acceptance criteria. The licensee's current June 21, 2010, relief request reiterates the November 14, 2009, request for relaxed repair weld acceptance criteria for all penetration nozzles at BVPS-2.

2.0 REGULATORY EVALUATION

The ISI of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3 components is to be performed in accordance with the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code and applicable editions and addenda as required by Section 50.55a(g) of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) except where specific written relief has been granted by the Commission. Pursuant to 10 CFR 50.55a(g)(4), throughout the service life of a pressurized-water cooled nuclear power facility, components which are classified 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. For BVPS-2, the ASME Code of Record for the third 10-year ISI interval, which began on August 29, 2008, and ends on August 28, 2018, is the 2001 Edition through the 2003 Addenda.

Alternatives to requirements may be authorized or relief granted by the NRC pursuant to 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), or 10 CFR 50.55a(g)(6)(i). 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; (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety; or (3) conformance is impractical for the facility.

3.0 TECHNICAL EVALUATION

3.1 System/Component Affected

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

3.2 Applicable Code Requirements

ASME Code, Section XI, 2001 Edition through 2003 Addenda, subparagraph IWA-4400 contains requirements for the removal of defects from and welded repairs performed on ASME Code components. For the removal or mitigation of defects by welding, ASME Code,

Section XI, 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-4553, which pertain to the removal of weld material defects prior to repair by welding.

3.3 Licensee's Proposed Alternative

3.3.1 Alternative NDE Acceptance Criteria

The following proposed alternative acceptance criteria supersede the acceptance criteria authorized by the NRC by letter dated October 6, 2009.

Nondestructive surface examination acceptance criteria of the original construction code shall be used for that part of the repair overlay that extends past the toe of the original alloy 600 J-groove weld. Specifically, the criteria, as detailed in ASME Code, Section III, 1971 Edition, Summer 1972 addenda (original construction code) and presented in Section 3.0, Applicable Code Requirements, shall be used. Nondestructive surface examination acceptance criteria for the remainder of the weld overlay shall be "PT [penetrant test] White."

The proposed alternative acceptance criteria applies to the post repair PT referred to in Section 5.2, Item 5, of Request 2-TYP-3-RV-03, and is to be used in lieu of establishing "PT White" conditions on the entire surface of the weld overlay repairs.

3.3.2 Alternative Filler Metal

The following proposed alternative filler metal is to be applied in addition to the weld repair method authorized by the NRC by letter dated October 6, 2009.

Prior to application of three alloy 52M repair weld layers on the clad surface, a minimum of three passes (one layer) of alloy ER309L shall be installed at the periphery of the weld overlay (at the repair-to-clad interface).

3.4 Licensee's Basis for Relief

Request 2-TYP-3-RV-03, as supplemented, is identical to the prior relief request, Request 2-TYP-3-RV-01, which was approved on October 6, 2009, with the exception of the currently requested final surface NDE acceptance criteria and alternative filler metal. The previously approved 10 CFR 50.55a request required the entire weld overlay repair surface to be examined by liquid dye PT with the acceptance criteria of "PT White," that is, no indications. The licensee's alternative is being requested because of a hardship to meet this requirement as significant radiation dose is estimated to be incurred to satisfy the "PT White" acceptance criteria versus the original construction code acceptance criteria.

The licensee noted that during the BVPS-2 refueling outage in the fall of 2009, ultrasonic examinations performed on penetrations 49 and 57 of the reactor vessel head revealed unacceptable indications in J-groove welds. These indications were subsequently repaired by the NRC approved embedded flaw weld overlay repair process described in 2-TYP-3-RV-01. The "PT White" acceptance criterion was not achieved initially. The licensee believed contamination from impurities in the original reactor vessel head cladding contributed to the indications at the weld periphery. The indications were predominantly located at the toe of the weld, and had an appearance typical of the solidification anomalies (that is, hot cracking) for which alloy 52M welding filler materials are known. In an effort to repair these indications, the licensee used successive grinding and re-welding, with additional surface examinations performed.

The licensee was not able to successfully remove all of the fusion boundary indications on penetration 57. Radiation exposure associated with a single weld overlay repair operation is approximately 10 rem. To obtain a weld overlay with no indications ("PT White") could require significant additional radiation exposure. It was estimated that an additional expenditure of approximately 10 rem of radiation exposure may have been needed to successfully remove the indications on penetration 57 in order to establish a "PT White" condition.

After review of the issue, the licensee proposed use of alternative filler metal to reduce the contaminant level and crack susceptibility in weld overlay repair material located over the stainless steel cladding, and thereby, reduce the number of indications.

In support of the proposal, the licensee stated that the purpose of the repair overlay welds is to embed and isolate identified flaws in the alloy 600 reactor vessel head penetration nozzle and/or its alloy 182 J-groove attachment weld. The repair overlay welds are not credited for providing structural strength to the original pressure boundary materials. The weld overlay repair extends a radial distance beyond the toe of the original J-groove welds by a minimum of a half inch. Repair weld overlays are a minimum of three layers in thickness.

The licensee finds it unlikely that indications resulting from the impurities in the original stainless steel cladding will be present in sufficient numbers or volume such that a path for communication of the reactor coolant back to the alloy 600 J-groove is possible. Additionally, the licensee finds the presence of these small indications in the weld repair deposit would have no propensity to cause crack extension. Therefore, the licensee concluded that use of construction code acceptance criteria, at the fusion line area of the repair where the weld overlay contacts the original reactor vessel head cladding, will ensure that the original susceptible material (alloy 600/182) will be isolated from the environment during operation. Further, crack growth and additional crack initiation resulting from PWSCC would be precluded.

The licensee also noted that successive post repair examinations of the J-groove welds repaired utilizing the embedded flaw weld overlay repair process will be conducted in accordance with 10 CFR 50.55a(g)(6)(ii)(D), which requires implementation of ASME Code Case N-729-1 with certain conditions. The requirements of ASME Code Case N-729-1 include successive post repair surface examinations of the weld overlay and successive post repair volumetric examinations of the alloy 600 reactor vessel head penetration nozzles.

By letter dated August 13, 2010, the licensee, in response to an NRC request for additional information, provided the current status of associated corrective actions, as well as, a root

cause analysis report associated with the difficulties in installation of a weld overlay repair on penetration nozzles 49 and 57 during the fall 2009 outage at BVPS-2. The root cause report contains proprietary and copyright information that is withheld from public disclosure in accordance with 10 CFR Section 2.390(b)(3). The licensee publicly noted that the report included a description of welding issues, welding conditions, power ratio information, and discussion of the use of alloy 52 versus alloy 52M weld wire.

Further, in an attachment to the August 13, 2010, letter, the licensee provided the status of corrective actions to prevent recurrence of the welding issues which may have lead to the increased radiological dose received during the fall 2009 outage repair activities. These actions include use of a barrier layer of ER309L filler material, updating weld parameters including voltage, amperage, travel speed and wire feed speed, and use of a full scale mockup to verify successful implementation of corrective actions and ensure an effective weld procedure for future application of this repair technique.

3.5 NRC Staff's Evaluation

The NRC staff finds that the licensee's estimate of an additional 10 rem of accumulated radiological dose per penetration repaired in order to be in compliance with the previously approved repair acceptance criteria under 2-TYP-3-RV-03 is a hardship.

Given this hardship, the NRC staff's review of the licensee's revised alternative of Request 2-TYP-3-RV-03 will be to ensure reasonable assurance of structural integrity under the frequency of inspection for reactor pressure vessel (RPV) upper head penetrations in accordance with 10 CFR 50.55a(g)(6)(ii)(D).

The purpose of the repair is to address PWSCC, which typically initiates in susceptible materials, such as alloy 600 material and alloy 82/182 weld materials, in areas of tensile stress and certain environmental conditions, such as higher temperatures and corrosive environments. The reactor vessel head penetrations and their associated J-groove attachment welds at BVPS-2 meet these conditions to be highly susceptible to PWSCC. The proposed repair technique isolates the susceptible material using a weld overlay of alloy 52M weld material, which is less susceptible to PWSCC. In order to ensure complete coverage of all high PWSCC susceptible material, the weld overlay extends an additional half inch beyond the outer ring of the original J-groove weld onto the stainless steel cladding covering the inside surface of the head. PWSCC is not considered a structural degradation mechanism for the stainless steel clad low alloy steel head.

The NRC staff reviewed the licensee's root cause report and corrective actions documented by letter dated August 13, 2010. The NRC staff finds that the licensee provided sufficient basis to assert that improved control of the root cause identified welding parameters will likely improve weld quality. Further, the NRC staff finds that use of a minimum of three passes (one layer) of alloy ER309L over the stainless steel cladding will assist in weld cleanliness and minimize cracking issues. This technique has been NRC approved and used in industry application of weld overlays on the outside of dissimilar metal butt welds in the primary coolant systems of several other plants. Therefore, the NRC staff finds the licensee's corrective actions provide reasonable assurance of a quality weld with minimal surface and subsurface fabrication issues.

The NRC staff reviewed the licensee's proposed alternative to reduce the surface NDE acceptance criteria for only that section of the alloy 52M weld over the stainless steel cladding of the RPV upper head. Current regulations under 10 CFR 50.55a(g)(6)(ii)(D) would require surface examination of this type of repair weld overlay each refueling outage. The NRC staff notes that due to PWSCC not being a structural concern for the RPV upper head materials for which this section of the overlay would be applied, the surface NDE acceptance criteria of the licensee's construction code will provide reasonable assurance of structural integrity of this section of the repair. Further, the NRC staff notes that due to the improved controls for weld quality and alloy 52M's resistance to PWSCC cracking, an allowed surface indication under the licensee's construction code would be highly unlikely to grow to a length to allow the susceptible weld or nozzle materials of a penetration nozzle to be affected between intervals of re-inspection. Therefore, the NRC staff finds, given the purpose of the repair and the licensee's effective corrective actions, the licensee's revised proposed alternative provides reasonable assurance of structural integrity.

The NRC staff finds that, given the licensee's identified hardship, requiring compliance with the current requirements of the previously approved relief request would cause an unnecessary burden on the licensee without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

Based on the above evaluation, the NRC staff has concluded that the licensee provided sufficient technical basis to find that compliance with the current requirements would cause an unnecessary burden on the licensee without a compensating increase in the level of quality and safety. Additionally, the NRC staff concluded that the licensee is in compliance with the ASME Code requirements and the proposed alternative provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the proposed alternative for the remainder of the current BVPS-2 10-year ISI interval, which ends August 28, 2018.

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: February 25, 2011

P. Harden

- 2 -

If you have any questions, please contact the Beaver Valley Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

/ra/

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosure:
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*see memo dated February 15, 2011

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