May 14, 2003

Mr. L. W. Pearce
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNITS 1 AND 2 - EVALUATION OF INSERVICE INSPECTION (ISI) RELIEF REQUEST BV3-RV-04 (TAC NOS. MB8172 AND MB8173)

Dear Mr. Pearce:

By letter dated March 28, 2003, as supplemented April 3, 4, and 7, 2003, First Energy Nuclear Operating Company (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," for repairs to the reactor pressure vessel (RPV) head control rod drive mechanism (CRDM) penetration nozzles for the Beaver Valley Power Station, Units 1 and 2 (BVPS-1 and 2). The licensee requested approval to use an alternative repair method utilizing an embedded flaw repair technique. Currently, the ASME Code, Section XI, subparagraphs IWA-4120 and IWA-4310, do not allow welding over or embedding an existing flaw.

The Nuclear Regulatory Commission (NRC) staff has completed its review of your relief request and the proposed alternative. As described in the enclosed safety evaluation, the NRC staff has authorized BV3-RV-04 for the remainder of the third and second 10-year ISI intervals, respectively, for BVPS-1 and 2, pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety.

This approval is conditional in that if the NRC staff finds that the crack growth formula in industry report MRP-55, “Material Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material (MRP-55),” Revision 1, is unacceptable, the licensee shall revise its analysis that justifies relief within 30 days after the NRC informs the licensee of an NRC-approved crack growth formula. If the licensee's revised analysis shows that the crack depth does exceed 75% of the CRDM nozzle wall thickness prior to the end of the current operating cycle, this relief is rescinded and the licensee shall, within 72 hours, submit to the NRC written justification for continued operation. If the revised analysis shows that the crack depth does exceed 75% of the CRDM nozzle wall thickness during the subsequent operating cycle, the licensee shall, within 30 days, submit the revised analysis for NRC review. If the revised analysis shows that the crack growth acceptance criteria are not exceeded during either the current operating cycle or the subsequent operating cycle, the licensee shall, within 30 days, submit a letter to the NRC confirming that its analysis has been revised.

This relief was granted verbally on April 18, 2003, at approximately 3:15 p.m. during a telephone conference call with members of your staff upon good cause shown and in consideration of the impact that might occur as the NRC staff was unable to process your request prior to the scheduled change to Mode 4 operation for BVPS-1.

If you have any questions regarding this approval, please contact the BVPS-1and 2 Project Manager, Mr. Timothy G. Colburn, at (301) 415-1402.

Sincerely,

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosure: Safety Evaluation

cc w/encl: See next page

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ACCESSION NO. ML031340697

*no major changes made
**See previous concurrence

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. BV3-RV-04

FOR REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

BEAVER VALLEY POWER STATION, UNITS 1 AND 2

FIRST ENERGY NUCLEAR OPERATING COMPANY

DOCKET NUMBERS 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated March 28, 2003, as supplemented April 3, 4, and 7, 2003, First Energy Nuclear Operating Company (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code or Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," for repairs to the reactor pressure vessel (RPV) head control rod drive mechanism (CRDM) penetration nozzles for the Beaver Valley Power Station, Units 1 and 2 (BVPS-1 and 2). The licensee requested approval to use an alternative repair method utilizing an embedded flaw repair technique. Currently, the ASME Code, Section XI, subparagraphs IWA-4120 and IWA-4310, do not allow welding over or embedding an existing flaw.

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of ASME Code, Class 1, 2, and 3 components is to be performed in accordance with the ASME Code, Section XI, and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) of 10 CFR states, in part, that alternatives to the requirements of paragraph (g) may be used when authorized by the Nuclear Regulatory Commission (NRC) if the applicant demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code, Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the 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. The 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 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ISI Code of record for the BVPS-1 and 2 third and second 10-year ISI intervals, respectively, is the 1989 Edition of Section XI of the ASME Code with no Addenda.

3.0 TECHNICAL EVALUATION

3.1 Code Requirements for which Relief is Requested

The ASME Code, Section XI, subparagraph IWA-4120, 1989 Edition, 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. Later Editions and Addenda of the Construction Code or of Section III, either in their entirety or portions thereof, and Code Cases may be used.

The ASME Code, Section XI, subparagraph IWA-4310, "Defect Removal Procedure," states, in part, that defects shall be removed or reduced in size in accordance with this Article. The ASME Code, Section XI, subparagraphs IWA-4120 and IWA-4310, do not allow welding over or embedding an existing flaw.

3.2 Licensee's Proposed Alternative to Code

The licensee proposes that any flaws requiring repair that are identified on reactor vessel head penetrations (VHP) and on the J-groove attachment welds will be embedded with a weld overlay.

For an inside diameter (ID) repair, an unacceptable axial flaw will be excavated or partially excavated to a depth no greater than 0.125 inches using the electric discharge machining process. Either an ultrasonic testing (UT) or eddy current testing (ET) examination of the excavated area will be performed to ensure the entire flaw length is captured. An Alloy 52 weldment will be applied to fill the excavation then the finished weld will be examined by dye penetrant testing (PT), UT or ET for acceptability. If an unacceptable ID circumferential flaw is detected, the flaw will either be repaired in accordance with existing Code requirements, or will be partially excavated to reduce the flaw to an acceptable size, examined by UT or ET, overlaid with Alloy 52, and examined by PT, UT or ET as described above.

For an outside diameter (OD) and J-groove weld repair, an unacceptable axial or circumferential flaw in a tube below a J-groove attachment weld will be sealed off with Alloy 52 weldment. This will be performed without excavation of the flaws since CRDM clearances are not a concern. Unacceptable radial flaws on the J-groove attachment weld will be sealed off with a 360°overlay of Alloy 52 covering the entire weld without performing excavation of the flaw. Unacceptable axial tube flaws extending into the J-groove attachment weld will be sealed with Alloy 52 and the entire J-groove attachment weld will be overlaid with Alloy 52 to embed the axial crack in the seal weld of the VHP. Finally, all the OD and J-groove repair welds will be examined by PT, UT or ET. The embedded flaw repair weld will be three layers thick for application to the J-groove attachment welds and at least two layers thick to base metal locations. For embedded flaw repairs involving flaws in the J-groove weld, the licensee will UT the OD of the penetration immediately above the weld.

No attempt will be made by the licensee to embed an OD circumferential flaw above the J-groove weld. Whenever an embedded flaw repair is planned for a circumferential flaw or J-groove weld repair, NRC notification will be made.

3.3 Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed repair alternative is requested by the licensee since the alternative provides an acceptable level of quality and safety due to the following reasons:

1. The licensee stated that as long as a Primary Water Stress Corrosion Cracking (PWSCC) flaw remains isolated from the primary water (PW) environment, it cannot propagate. Since Alloy 52 weldment is considered highly resistant to PWSCC, a new PWSCC flaw cannot initiate and grow through the Alloy 52 weld overlay to reconnect the PW environment with the embedded flaw. Structural integrity of the affect VHP J-groove attachment weld will be maintained by the remaining unflawed portion of the weld.

2. The licensee stated that the residual stresses produced by the embedded flaw technique have been measured and found to be relatively low. This was documented in the attachment to a letter from E. E. Fitzpatrick, Indiana Michigan Power Company, to the staff, "Reactor-Vessel Head Penetration Alternated Repair Techniques," dated March 12, 1996. The low residual stresses indicate that no new flaws will initiate and grow in the area adjacent to the repair weld.

3. The licensee stated that there are no other known mechanisms for significant flaw propagation in this region since cyclic fatigue loading is negligible.

4. The licensee stated that as a precedent the NRC approved a similar alternative for North Anna Power Station, Unit 2, on January 23, 2003. Additionally, the NRC previously approved a similar alternative for D.C. Cook, Units 1 and 2, on April 9, 1996. Although the alternative was applied to the VHP tube base metal rather than VHP welds, both alternatives used an embedded flaw repair technique.

Pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed repair alternative is requested by the licensee since the implementation of a more traditional, ASME Code, repair approach would be a hardship due to the following reasons:

1. The licensee stated that the dose estimate to perform an ASME Code-required repair is expected to be significantly higher than the dose estimate for the embedded flaw repair. Based on current industry experience, the licensee estimated that the dose may exceed 20 person-rem, while the person-rem estimate for the embedded flaw repair is approximately an order of magnitude lower.

2. The licensee stated that implementation of a traditional repair can be time consuming, and also involves increased risk, as the amount of structural deformation increases as the amount of deposited weld metal increases.

3.4 Evaluation

The 1989 Edition of ASME Code, Section XI, subparagraph IWA-4120, states that repairs shall be performed in accordance with the Owner's Design Specification and the original Construction Code of the component or system. Later editions and addenda of the Construction Code or of Section III, either in their entirety or portions thereof, and Code Cases may be used. The 1989 Edition of ASME Code, Section III, requires that prior to welding, the repair excavation would require examination per paragraph NB-4453 with the acceptance criteria of NB-5351 and NB-5352. In neither case is it permissible to weld over or embed an existing flaw. The alternative proposed by the licensee is to embed the defect with a weld overlay rather than excavate per the requirements of ASME Code, Section III.

The NRC staff concurs with the licensee's statement that conventional ASME Code repair methodology results in significant dose accumulation to personnel. Excavating and welding the resultant cavity has required that personnel maintain contact with the activated VHP as experienced at North Anna Power Station. High dose accumulation is a significant hardship, warranting that the embedded flaw methodology be considered as an alternative to the Construction Code or ASME Code, Section III, requirements. By letter dated February 5, 1996, the NRC staff approved the embedded flaw repair technique for CRDM ID flaws that exceeded the flaw acceptance criteria stated in Westinghouse Topical Report, WCAP-14024, "Inspection Plan Guidelines for Industry/Plant Inspection of Reactor Vessel Closure Head Penetration Tubes," Revision 1, and Westinghouse Topical Report, WCAP-14519, "RV Closure Head Penetration ID Weld Overlay Repair." In the conclusion section of the safety evaluation for WCAP-14519, the staff stated: "The NRC staff finds that the Westinghouse overlay process provides an acceptable alternative to the ASME Code in this Topical Report since it provides sufficient wall thickness such that leakage will not occur through the VHP wall during normal operation. Licensees may reference the Topical Report and this safety evaluation when requesting permission to use the alternative to the Code." The remainder of the safety evaluation will determine whether the alternative repair performed under the embedded flaw methodology provides an acceptable level of quality and safety for embedded flaw repairs applied to the exterior of the CRDM nozzle material below and over the J-groove weld area. Based on the above discussions, the NRC staff concludes that the embedded flaw process provides an acceptable level of quality and safety for repairs to the ID of CRDM nozzles.

In its supplemental letter dated April 4, 2003, the licensee submitted Westinghouse document, "The Embedded Flaw Process for Repair of Reactor Vessel Head Penetrations and Its Application at North Anna Unit 2," WCAP-15986, March 2003. This topical report discusses the effectiveness of the embedded flaw process including overlay of the J-groove weld area at North Anna Power Station, Unit 2, and service life of weld repairs in a variety of environments for the past 10 years using the embedded flaw methodology.

The embedded flaw process involves the deposition of at least two layers of Alloy 52 weld metal on Alloy 600 material nozzles and three layers on J-groove welds to isolate existing flaws and susceptible material from the primary water environment. Typically the NRC staff requires that the licensee submit site-specific information in order to evaluate the licensee's conclusions for crack growth rate estimates based on the applicable unit 's stress analyses, operating times and effective degradation years. In this instance, the service life of repairs performed to date will be considered by the NRC staff as part of the licensee's rationale because of the generic application of an Alloy 52 weld repair to Alloy 600 material.

WCAP-15986 provides a tabular listing of weld metal applications for Westinghouse-designed components in Section 4.2 entitled "Service Experience." This table lists 25 examples where repairs have been performed using Alloys 52/152 on operating plants since 1994 where no service-related failures have been discovered after repair. The repairs were conducted on steam generator nozzle welds, tubesheet cladding, divider plate-tubesheet welds, canopy seal overlays, pressurizer nozzle repairs, etc.

Furthermore, the service experience with D.C. Cook, Unit 2, CRDM penetration no. 75 nozzle repair was cited as an example. This plant had repaired a penetration with the embedded flaw repair process in 1996 and ET examined after 6 years of service in January, 2002. The ET results showed that no cracking had extended beyond the repair and by lack of any other results indicate that the cracking has not propagated through the Alloy 52 repair material after 6 years service. This is consistent with the performance and industry experience with full structural overlays on boiling water reactor (BWR) austenitic piping over through-wall Intergranular Stress Corrosion Cracking (IGSCC). The NRC staff concludes that sufficient field results exist, in a variety of environments and applications to date, to indicate that cracking does not extend beyond the boundary of the embedded flaw weld repair nor through the Alloy 52 repair, and is therefore, acceptable.

The licensee stated that the thermal expansion properties of Alloy 52 weld metal are not listed in the ASME Code, however, the properties of the equivalent base metal, Alloy 690 should be considered. For Alloy 690, the thermal expansion coefficient at 600°F is 8.2 E-6in/in/°F as found in Section II, Part D of the ASME Code. The Alloy 600 nozzle base metal to be repaired with an overlay has a coefficient of thermal expansion of 7.8 E-6 in/in/°F. The licensee concluded that this small difference in thermal expansion causes the weld metal to contract more than the base metal when it cools, producing a compressive stress on the Alloy 600 tube. The NRC staff reviewed the data provided in WCAP-13998 to determine if the data supported the conclusion that differing thermal expansions leave compressive stresses on the opposing surface. Measurements of the OD of a penetration tube mockup were recorded before and after a 360°, Alloy 52 weld overlay was deposited on the ID, with deposited weld depths of 0.09, 0.25, and 0.47 inches. In all but a few instances, Figures 6.1-1 through 6.1-8 of WCAP-13998 show the deformation of the OD resulting from weld shrinkage on the ID. The post weld measurements of the OD resulted in a decrease of the OD circumference of the penetration tube mockup, indicating compressive stresses were induced on the opposite side (OD) of the 360° overlay. This is consistent with industry experience when performing full structural overlays of austenitic material IGSCC. The NRC staff concludes from the data provided in WCAP-13998 that the embedded flaw process weld overlay procedure will leave compressive stresses on the opposite surface. Compressive stresses will reduce the driving force for cracking and are, therefore, acceptable to the NRC staff.

The licensee stated that there are no other known mechanisms for significant flaw propagation in this region since cyclic fatigue loading is negligible. In its supplemental letter dated April 7, 2003, the licensee indicated that the Fatigue Usage Factor for the CRDM region of the BVPS-1 and 2 CRDM housings was determined to be 0.0972 and 0.138 respectively, compared to the

ASME Code allowable value of 1.0. The NRC staff concludes that the information provided by the licensee indicates that fatigue driven crack growth is not a mechanism for further crack growth after the embedded flaw repair is performed and is, therefore, acceptable.

In its supplemental letter dated April 7, 2003, the licensee indicated that post repair NDE would be performed on the weld repair areas. The licensee stated that these post repair wetted surfaces will be examined by PT and the portion of the tube below the weld will be examined using both ET and UT. The NRC staff finds that this provides reasonable assurance of structurally sound repairs and is therefore, acceptable.

NRC Order EA-03-009, issued on February 11, 2003, requires pressurized water reactor licensees with CRDM cracking to perform non-destructive examination (NDE) each outage to monitor the integrity of the penetrations. In its supplemental letter dated April 7, 2003, the licensee provided BVPS-1 crack growth estimates for ID cracks to grow to 75% of the CRDM nozzle wall thickness. The crack growth calculations used a methodology consistent with the recently proposed ASME Code, Section XI, flaw evaluation approach using a crack growth rate (CGR) formula as provided in the Electric Power Research Institute's (EPRI) Material Reliability Program (MRP) Report, MRP-55, "Material Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick Wall Alloy 600 Material (MRP-55)," Revision 1. The NRC staff has only completed a preliminary review of MRP-55, but finds that the CGR used by the licensee is acceptable in the interim until the NRC staff completes its review of MRP-55. The application of the CGR formula is considered acceptable at this time because it does not consider the slower growth caused by the compressive stresses on the ID caused by the weld overlay of the OD. For BVPS-1, the most limiting penetration is the 42.6° penetration (as the nozzle relates to the head curvature) which would require 4.2 years of operation for the flaw to grow to 75% of the CRDM wall thickness. The NRC staff concludes that the performance of NDE each outage per Order EA-03-009, along with the use of the interim acceptable MRP-55 CGR model, will detect a flaw prior to growth to the Code allowable limit and is, therefore, acceptable.

4.0 CONCLUSION

From the discussions above, the NRC staff conclusions are as follows:

1. The embedded flaw process for ID flaws has been acceptable to the NRC staff since 1996.

2. Nearly 10 years of service experience with Alloy 52/152 embedded flaw repairs indicate satisfactory performance in a variety of environments, including CRDM materials.

3. Data provided from a previously approved Topical Report, WCAP-14519, indicates that embedded flaw repairs to the OD and J-groove weld will place the flaw in compression which is a desirable state.

4. Both post-repair NDE and the periodic NDE required by NRC Order EA-03-009 will monitor the performance of the repair and detect any additional cracking prior to the flaw exceeding Code acceptance criteria.

5. Licensee-provided data indicates that the potential for fatigue driven failure is low.

Based on the discussion above, the NRC staff concludes that the alternative to perform the embedded flaw repair technique proposed in Relief Request No. BV-3-RV-04, at BVPS-1 and 2, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the proposed alternative for the third and second 10-year ISI intervals at BVPS-1 and 2 , respectively. 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.

This approval has one condition. If the NRC staff finds that the crack growth formula in industry report MRP-55 is unacceptable, the licensee shall revise its analysis that justifies relief within 30 days after the NRC informs the licensee of an NRC-approved crack growth formula. If the licensee's revised analysis shows that the crack depth does exceed 75% of the CRDM nozzle wall thickness prior to the end of the current operating cycle, this relief is rescinded and the licensee shall, within 72 hours, submit to the NRC written justification for continued operation. If the revised analysis shows that the crack depth does exceed 75% of the CRDM nozzle wall thickness during the subsequent operating cycle, the licensee shall, within 30 days, submit the revised analysis for NRC review. If the revised analysis shows that the crack growth acceptance criteria are not exceeded during either the current operating cycle or the subsequent operating cycle, the licensee shall, within 30 days, submit a letter to the NRC confirming that its analysis has been revised.

Principal Contributor: T. Steingass

Date: May 14, 2003