October 2, 2006

Mr. James H. Lash
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 NOS. 1 AND 2 (BVPS-1 AND 2), INSERVICE INSPECTION (ISI) PROGRAM, ALTERNATIVE EXAMINATION OF REACTOR COOLANT PIPE WELDS, REQUEST FOR RELIEF NO. BV3-RV-2 (TAC NOS. MD1137 AND MD1138)

Dear Mr. Lash:

By letter dated April 7, 2006, FirstEnergy Nuclear Operating Company (the licensee) submitted relief request BV3-RV-2, for the third 10-year ISI interval at BVPS-1 and for the second 10-year ISI interval at BVPS-2. The licensee requested Nuclear Regulatory Commission (NRC) approval to use an alternative examination of reactor coolant pipe welds.

The NRC staff has completed its review and evaluated the information regarding the relief request for BVPS-1 and 2. The results are provided in the enclosed safety evaluation. The staff concludes that an acceptable level of quality and safety will be maintained upon implementation of the licensee's proposed alternative examination and therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative examination is authorized for the remainder of the third 10-year ISI interval at BVPS-1, and for the remainder of the second 10-year ISI interval at BVPS-2.

If you have any questions, please contact your NRC Project Manager, Mr. Timothy G. Colburn, at 301-415-1402.

Sincerely,

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
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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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BV3-RV-2, INSERVICE INSPECTION (ISI) RELIEF REQUEST

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated April 7, 2006, FirstEnergy Nuclear Operating Company (the licensee) requested, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i), that the Nuclear Regulatory Commission (NRC) approve relief request BV3-RV-2 which pertains to the examination of certain reactor coolant pipe welds during the third 10-year ISI interval at BVPS-1 and the second 10-year ISI interval at BVPS-2. American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, 1989 Edition, no Addenda is the ISI Code of record for the BVPS-1 third 10-year ISI Interval and the BVPS-2 second 10-year ISI interval.

2.0 REGULATORY EVALUATION

10 CFR 50.55a(g) specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).
10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (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.
10 CFR 50.55a(g)(5)(iii) states that if the licensee has determined that conformance with certain Code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in Section 50.4, information to support the determinations.

3.0 TECHNICAL EVALUATION

The information provided by the licensee in support of the request has been evaluated by the NRC staff and the bases for disposition are documented below.

Enclosure

3.1 Licensee's Evaluation

3.1.1 ASME Code Component Affected

BVPS-1 Reactor Coolant Pipe Welds: six reactor vessel nozzle to loop piping welds. The stainless steel welds join the carbon steel nozzle to cast stainless steel piping material A351, Gr. CF8M.

BVPS-2 Reactor Coolant Pipe Welds: six reactor vessel nozzle to safe-end welds and six safe-end to loop piping welds. Inconel welds join the carbon steel nozzle to 316 stainless steel safe-end. Stainless steel welds join the 316 stainless steel safe-end to the cast stainless piping material SA351, Gr. CF8A.

3.1.2 Applicable Code Edition and Addenda

ASME Code, Section XI, 1989 Edition, no Addenda.

ASME Code, Section XI, 1995 Edition, 1996 Addenda (Appendix VIII, Supplements 2 and 10, as required by 10 CFR 50.55a(g)(6)(ii)(C).

3.1.3 Applicable Code Requirements

Examination Category R-A, Item R1.11 (RI-ISI Program categorization), formerly Category B-F, Item Number B5.10 (nozzle to safe-end/nozzle to pipe welds) and Category B-J, Item Number B9.11 (safe-end to piping welds, BVPS-2 only) specify volumetric examination. The volumetric examination is to be conducted in accordance with Appendix VIII, Supplements 2 and 10, of the 1995 Edition with 1996 Addenda per 10 CFR 50.55a(g)(6)(ii)(C).

3.1.4 Reason for Request

ASME Code Case N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1," was approved by the ASME Main Committee on May 21, 2003, but has not been published in Regulatory Guide 1.147.
Code Case N-696 addresses the combined qualification for Supplements 2, 3, and 10 when examinations are conducted for the inside surface. Although examination vendors have been qualified for detection and length sizing on these welds, no examination vendor has yet to meet the established root mean squared error (RMSE) criteria for depth sizing. The licensee's contracted vendor has demonstrated ability for depth sizing qualification with an RMSE of 0.245 inches instead of the 0.125 inches required by the code case.

3.1.5 Proposed Alternative and Basis for Use

The licensee proposes to use Code Case N-696 with an RMSE of 0.245 inches instead of the 0.125 inches specified for depth sizing in the code case. In the event an indication is detected that requires depth sizing, the 0.120-inch difference between the required RMSE and the demonstrated RMSE (0.245 inches minus 0.125 inches = 0.120 inches) will be added to the estimated through-wall extent for comparison with applicable acceptance criteria. If the examination vendor demonstrates an improved depth sizing RMSE prior to the examination,

then the difference between the demonstrated RMSE and the required 0.125-inch RMSE requirement, if any, will be added to the measured value for comparison with applicable acceptance criteria.

The activities included in this relief are subject to third party review by the Authorized Nuclear Inservice Inspector.

The licensee proposes the specified alternative in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative assures that the subject reactor coolant piping welds will be fully examined by procedures, personnel, and equipment qualified by demonstration to Code Case N-696 requirements in all aspects except depth sizing. For depth sizing, the difference between the required and demonstrated sizing tolerance will be added to any flaw required to be sized to compensate for the variance in depth sizing as demonstrated by the examination vendor. Therefore, the licensee states that the use of Code Case N-696, with the stated compensation for depth sizing, provides an acceptable level of quality and safety.

3.1.6 Duration of the Proposed Alternative

The proposed alternative is requested for the remainder of the third 10-year ISI interval at BVPS-1 and for the remainder of the second 10-year ISI interval at BVPS-2.

3.2 NRC Staff's Evaluation

Supplement 10 of Appendix VIII to ASME Code, Section XI, requires that examination procedures, equipment, and personnel meet specific criteria for flaw depth-sizing accuracy. The Code specifies that the maximum error of flaw depth measurements, as compared to the true flaw depths, must be less than or equal to 0.125-inch RMSE.

The licensee proposed to use Code Case N-696 with an RMSE of 0.245 inches instead of the 0.125 inches specified for depth sizing in the code case. In the event that an indication is detected that requires depth sizing, the 0.120 inch difference between the required RMSE and the demonstrated RMSE (0.245 inches minus 0.125 inches = 0.120 inches) will be added to the estimated through-wall extent for comparison with applicable acceptance criteria.

The nuclear industry is in the process of qualifying personnel in accordance with Supplements 2 and 10 requirements, as implemented through the PDI program. However, personnel have been unsuccessful in achieving the Code-required RMSE value for flaw depth-sizing demonstrations performed from the inside surface of a pipe weld. At this time, achieving an RMSE value of 0.125 inches is impractical since no vendor has been able to comply with the Code-required value. The performance of the vendor with an RMSE of 0.245 inches, represents the current achievable state of practice for through-wall sizing from the inside surface of the reactor vessel nozzle. As a result, the licensee is proposing to use a depth-sizing criterion of 0.245 inches to size any detected flaw during the examination of the subject safe end-to-pipe welds. The licensee also proposes to add the difference (0.120 inches) between the Code-required RMSE (0.125 inches) and the demonstrated accuracy (0.245 inches) to the measurements acquired from flaw sizing.

The Code requirement of 0.125-inch RMSE for acceptable flaw depth-sizing error is for examinations of nozzle-to-safe end dissimilar metal welds and safe end-to-piping similar metal welds from the inside surfaces of the components using automated equipment. The demonstrated accuracy for these welds during performance qualifications was 0.245-inches. The licensee's practice of adding the difference between the Code-required RMSE and the demonstrated accuracy to the measurements acquired from sizing any detected flaws, in addition to the use of the acceptance standards specified in Section IWB-3500 of the Code, provides an acceptable level of quality and safety.

4.0. CONCLUSION

The licensee has provided an acceptable alternative to ASME Code, Section XI, requirements for the requested relief. The NRC staff has determined that the proposed alternative provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the remainder of the third 10-year ISI interval at BVPS-1 and for the remainder of the second 10-year ISI interval at BVPS-2. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: G. Georgiev

Date: October 2, 2006

Beaver Valley Power Station, Unit Nos. 1 and 2

cc:

Gary R. Leidich
President and Chief Nuclear Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-19
76 South Main Street
Akron, OH 44308

Joseph J. Hagan
Senior Vice Presdient of Operations
and Chief Operating Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Danny L. Pace
Senior Vice President, Fleet Engineering
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Jeannie M. Rinckel
Vice President, Fleet Oversight
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

David W. Jenkins, Attorney
FirstEnergy Corporation
Mail Stop A-GO-18
76 South Main Street
Akron, OH 44308

Manager, Fleet Licensing
FirstEnergy Nuclear Operating Company
Mail Stop A-GHE-107
395 Ghent Road
Akron, OH 44333

James H. Lash
Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

Lew W. Myers
Executive Vice President, Special Projects
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SGRP
P.O. Box 4, Route 168
Shippingport, PA 15077

Manager, Site Regulatory Compliance
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-A
P.O. Box 4, Route 168
Shippingport, PA 15077

Commissioner James R. Lewis
West Virginia Division of Labor
749-B, Building No. 6
Capitol Complex
Charleston, WV 25305

Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency
Management Agency
2605 Interstate Dr.
Harrisburg, PA 17110-9364

Ohio EPA-DERR
ATTN: Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

cc:

Dr. Judith Johnsrud
Environmental Coalition on Nuclear Power
Sierra Club
433 Orlando Avenue
State College, PA 16803

Director
Bureau of Radiation Protection
Pennsylvania Department of
  Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Mayor of the Borough of Shippingport
P.O. Box 3
Shippingport, PA 15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 298
Shippingport, PA 15077