FENOC

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L-13-029

10 CFR 50.55a

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
10 CFR 50.55a Alternative Examination Request for Reactor Vessel Safe-End Welds

In accordance with 10 CFR 50.55a, Nuclear Regulatory Commission (NRC) review and approval is requested for a proposed alternative to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements associated with volumetric examinations of Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel nozzle to safe-end dissimilar metal welds.

The affected components, the applicable ASME Code requirements, a description of the proposed alternative and basis for use is provided in the Enclosure. The alternative is proposed for use during the remainder of the current BVPS-2 10-year inservice inspection interval, which began on August 29, 2008.

The proposed alternative is to be implemented prior to the spring 2014 BVPS-2 refueling outage. Therefore, FirstEnergy Nuclear Operating Company requests approval of the proposed alternative by March 31, 2014.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

Sincerely,

Paul A. Harden

Enclosure:
10 CFR 50.55a Request Number: 2-TYP-3-RVSE-2

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Proposed Alternative
in Accordance with 10 CFR 50.55a(g)(5)(iii)

--Inservice Inspection Impracticality--

1.0 ASME Code Components Affected

Code Class: Class 1
System: Reactor Coolant System (RCS)

In accordance with 10 CFR 50.55a(g)(6)(ii)(F)(1), the reactor vessel nozzle to safe-end dissimilar metal welds listed below are subject to volumetric examinations in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds..."

Component	Description of Weld	Inspection Item
2RCS*REV21-N-24	Reactor Vessel Hot Leg Nozzle to Safe-End	A-2
2RCS*REV21-N-26	Reactor Vessel Hot Leg Nozzle to Safe-End	A-2
2RCS*REV21-N-28	Reactor Vessel Hot Leg Nozzle to Safe-End	A-2
2RCS*REV21-N-23	Reactor Vessel Cold Leg Nozzle to Safe-End	B
2RCS*REV21-N-25	Reactor Vessel Cold Leg Nozzle to Safe-End	B
2RCS*REV21-N-27	Reactor Vessel Cold Leg Nozzle to Safe-End	B

As determined by FirstEnergy Nuclear Operating Company (FENOC), additional welds that may be volumetrically examined concurrent with the above listed welds are the reactor vessel safe-end to pipe austenitic welds listed below. These welds are part of the risk-informed inservice inspection program and are listed in the Examination Schedules as Examination Category R-A, Item R1.11.

Component	Description of Weld	Inspection Item
2RCS*001-F01	Reactor Vessel Hot Leg Safe-End to Pipe	R1.11
2RCS*004-F01	Reactor Vessel Hot Leg Safe-End to Pipe	R1.11
2RCS*007-F01	Reactor Vessel Hot Leg Safe-End to Pipe	R1.11
2RCS*003-F04	Reactor Vessel Cold Leg Safe-End to Pipe	R1.11
2RCS*006-F04	Reactor Vessel Cold Leg Safe-End to Pipe	R1.11
2RCS*009-F04	Reactor Vessel Cold Leg Safe-End to Pipe	R1.11

2.0 Applicable Code Edition and Addenda

Beaver Valley Power Station Unit No. 2 (BVPS-2) In-Service Inspection and Repair/Replacement Programs:	ASME Code Section XI, 2001 Edition through 2003 Addenda
BVPS-2 Ultrasonic (UT) Examination:	ASME Code Section XI, 2001 Edition with no Addenda and Appendix VIII Supplements

3.0 Applicable Code Requirement

The volumetric examinations are to be conducted in accordance with Appendix VIII, "Performance Demonstration Initiative [PDI] for Ultrasonic Examination System," Supplements 2 and 10 of the ASME Code Section XI, 2001 Edition. Alternatives to Appendix VIII, Supplements 2 and 10, are ASME Code Cases N-695 (Supplement 10) and N-696 (combined Supplements 2 and 10). These Code Cases were approved for use by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Table 1, "Acceptable Section XI Code Cases."

Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," Paragraph 3.3(c) states, "Examination procedures, equipment, and personnel are qualified for depth-sizing when the RMS [root mean square] error of the flaw depth measurements, as compared to the true flaw depths, do not exceed 0.125 in. (3 mm)."

Code Case N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted from the Inside Surface, Section XI, Division 1," Paragraph 3.3(d) states, "Supplement 2...examination procedures, equipment, and personnel are qualified for depth-sizing when the flaw depths estimated by ultrasonics, as compared with the true depths, do not exceed 0.125 in. (3mm) RMS, when they are combined with a successful Supplement 10 qualification."

4.0 Impracticality of Compliance

An Electric Power Research Institute (EPRI) letter dated March 8, 2012 states, "To date, no domestic or international vendor has met the applicable root mean square (RMS) error requirement specified in the ASME Code."

When examining from the inside diameter, the vendor attempts to meet the Supplement 10 (Code Case N-695) and combined Supplement 2 and 10 (Code Case N-696) required RMS error values for flaw depth sizing have been unsuccessful. Process enhancements including new delivery systems, new transducers, and software modifications have been implemented but have not achieved the desired improvements in performance. This result indicates the ASME Code acceptance requirement for flaw depth sizing is impractical for use with current inside diameter ultrasonic examination technology.

5.0 Burden Caused by Compliance

Compliance with the performance demonstration initiative qualification program without an alternative qualification requirement would necessitate significant modifications to the reactor coolant system welds. Alterations such as these may result in reduced structural integrity of the reactor coolant pressure boundary. Even with such modifications, the vendor depth sizing accuracy issue would not likely be fully addressed.

6.0 Proposed Alternative and Basis for Use

As approved for use by the NRC in Regulatory Guide 1.147, FENOC proposes to use ASME Code Case N-696 to perform a combined Supplements 2 and 10 qualification when examining the reactor vessel nozzle to safe-end dissimilar metal welds and reactor vessel safe-end to pipe austenitic welds. If only the reactor vessel nozzle to safe-end dissimilar metal welds are examined, FENOC proposes to use ASME Code Case N-695 to perform a Supplement 10 qualification. However, FENOC proposes using alternative RMS error depth sizing criteria as compared to the values stated in Code Cases N-695 and N-696.

The FENOC inside diameter examination vendor has demonstrated the ability to depth size flaw indications in dissimilar metal welds with an RMS error of 0.189 inch instead of the 0.125 inch RMS error required by Appendix VIII Supplement 10 (Code Case N-695) and the RMS error of 0.245 inch for the combined Appendix VIII Supplements 2 and 10 qualification (Code Case N-696). The difference between the 0.245 inch RMS error and the Code Case N-696 required 0.125 inch RMS error would be added to the flaw depths determined during actual sizing of flaws. FENOC also proposes that if only the reactor vessel nozzle to safe-end dissimilar metal welds are examined, the difference between the 0.189 inch RMS error and the Code Case N-695 required 0.125 inch RMS error would be added to the flaw depths determined during actual sizing of flaws. Figures 1 and 2 provide representative sketches of a reactor vessel outlet nozzle to safe-end weld and an inlet nozzle to safe-end weld.

If a flaw or flaws are detected and are measured as less than 50 percent through-wall in depth, adding the proposed correction factor of the RMS error minus 0.125 inches to the depths of any flaws is an acceptable alternative.

If a flaw or flaws are detected and they are measured as greater than 50 percent through-wall depth, and the flaws will be left in service without mitigation or repair, flaw evaluations will be submitted to the NRC for review, and receipt of NRC approval will be required prior to reactor startup. In addition to the information normally contained in flaw evaluations, this evaluation shall include:

a. Information concerning the mechanism that caused the crack
b. Information concerning the surface roughness/profile in the area of the pipe/weld required to perform the inspection
c. Information concerning areas in which the UT probe may lift off from the surface of the pipe/weld

The nominal diameters and thickness dimensions for the reactor vessel outlet and inlet nozzle to safe-end welds, as well as the safe-end to pipe welds, are provided in Table 1.

Table 1: Nominal Diameter and Wall Thickness Dimensions

Weld Description	Nominal Inside Diameter	Nominal Outside Diameter (OD)	Nominal Thickness
Reactor Vessel Outlet Nozzle to Safe-End	28.97"	34.22"	2.63"
Reactor Vessel Outlet Safe-End to Pipe	29.20"	34.22"	2.51"
Reactor Vessel Inlet Nozzle to Safe-End	27.47"	32.47"	2.50"
Reactor Vessel Inlet Safe-End to Pipe	27.70"	32.47"	2.39"

The BVPS-2 reactor vessel nozzle weld examinations would be completed using the inside diameter applied PDI-qualified UT equipment, personnel, procedures and techniques qualified by the vendor. During the nozzle weld examination process, inside diameter surface profile data would be recorded using an immersion UT process. Contour wedges that match the inside diameter contour will be utilized when the examination is performed. This display is interactive with the actual UT data such that key flaw characterization information such as flaw depth sizing and flaw location can be compensated for by using the surface profile directly underneath the transducer. This profilometry software has the ability to calculate the areas where the water path under the transducer is greater than 1/32 inch; this information is used to calculate examination volume coverage where detection scans are limited. The 1/32 inch value is generally considered for OD examinations but offers a conservative reference for the inside diameter examination. Based on BVPS-2 exams performed in 2008 and 2012, the surface conditions of these nozzle to safe-end dissimilar metal welds are relatively smooth, and therefore, no lack of coverage would be expected.

There are no limitations for axial scans for circumferential flaws, but the same profilometry software discussed earlier is used to detect areas where limitations for UT exist during the circumferential scans for axial flaws.

FENOC's examination vendor has participated in three non-ASME Code required performance demonstrations associated with depth sizing of planar flaws in dissimilar metal welds. The demonstrated techniques were conducted from the inside diameter surface. Each of these demonstrations used UT test procedures and equipment similar to those to be applied for the BVPS-2 weld examinations. Summary information on these demonstrations was provided in our December 27, 2011 letter (Accession No. ML113620646).

As stated by EPRI, no domestic or international vendor has met the applicable RMS error requirement specified in the ASME Code. The proposed alternative assures that the subject welds would be fully examined by vendor procedures, equipment and personnel qualified by demonstration in all aspects except depth sizing. For depth sizing, the proposed addition of the numeric difference between the required and

demonstrated achievable sizing tolerance to any flaw that is required to be sized compensates for the potential variation. FENOC has determined that the proposed alternative provides an acceptable level of quality and safety, pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii).

7.0 Duration of Proposed Alternative

The proposed alternative shall be utilized during the remainder of the BVPS-2 third 10-year inservice inspection interval, which began on August 29, 2008, and is currently scheduled to expire on August 28, 2018.

8.0 Precedent

1. NRC Safety Evaluation Regarding: Beaver Valley Power Station, Unit No. 2 – Request for Relief Relating to Reactor Vessel Nozzle Welds (TAC No. ME7770), July 18, 2012 (Accession No. ML12188A110).

2. NRC Safety Evaluation Regarding: McGuire Nuclear Station, Unit 2, Proposed Relief Request 12-MN-003 (TAC No. ME8712), September 24, 2012 (Accession No. ML12258A363).

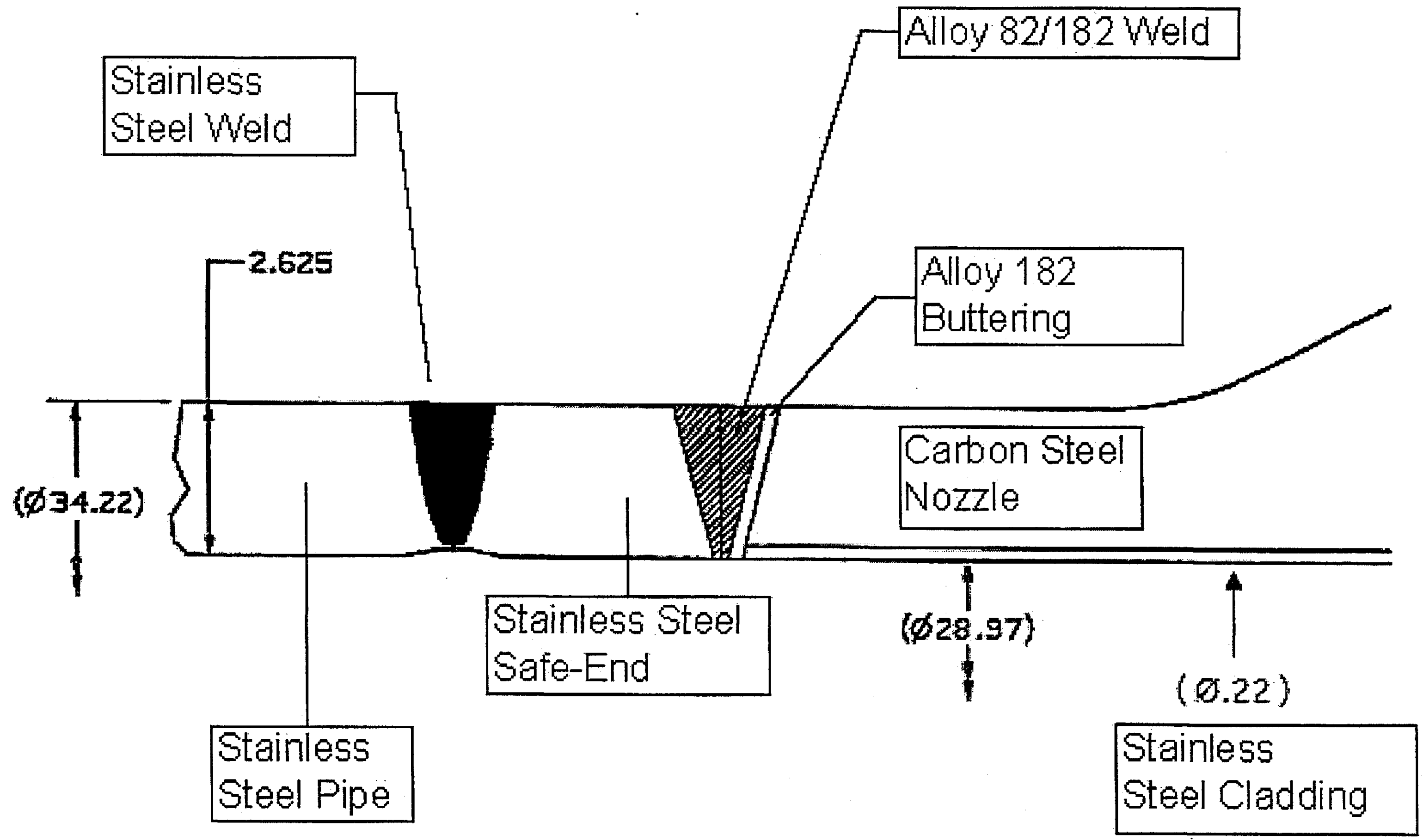

Figure 1: BVPS-2 Reactor Vessel Outlet Nozzle

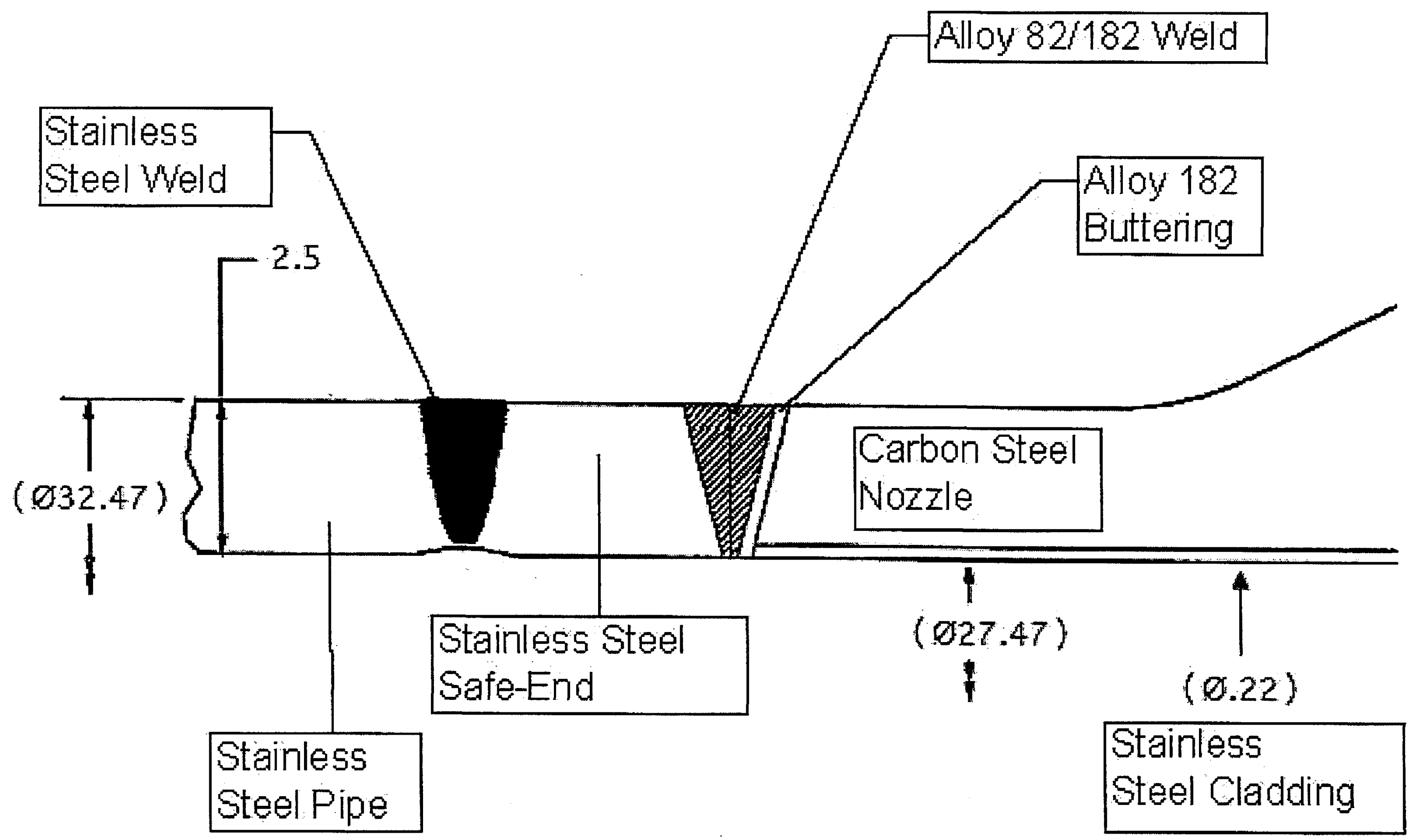

Figure 2: BVPS-2 Reactor Vessel Inlet Nozzle