

Mark B. Bezilla
Vice President440-280-5382
Fax: 440-280-8029January 24, 2011
L-10-246

10 CFR 50.55a

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001**SUBJECT:**Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
10 CFR 50.55a Requests in Support of the Third 10-Year In-Service Inspection Interval

In accordance with 10 CFR 50.55a, the FirstEnergy Nuclear Operating Company (FENOC) hereby submits the following requests, in support of the Third 10-Year In-Service Inspection (ISI) Interval for the Perry Nuclear Power Plant (PNPP).

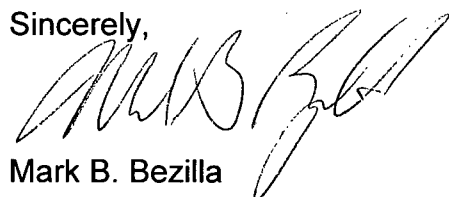
- One notification of impracticality in accordance with 10 CFR 50.55a(g)(5)(iii) (Enclosure A);
- Four proposed alternatives providing acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i) (Enclosures B, G, H, and I); and
- Four proposed alternatives for hardship or unusual difficulty without compensating increase in level of quality or safety in accordance with 10 CFR 50.55a(a)(3)(ii) (Enclosures C, D, E, and F).

As applicable for their respective relief requests, the Enclosures identify impracticality of compliance, the proposed alternatives, the affected components, the applicable code requirements, the reason for the requests, and the basis for use.

The relief is requested for the PNPP during the third 10-year ISI interval and is needed to finalize the scope of the spring 2013 refueling outage. Therefore, FENOC is requesting approval of the proposed 10 CFR 50.55a requests by January 31, 2012.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

Sincerely,



Mark B. Bezilla

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NLR

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Enclosures:

- A. Perry Nuclear Power Plant, 10 CFR 50.55a IR-001, Revision 3
- B. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-009, Revision 2
- C. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-012, Revision 3
- D. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-013, Revision 2
- E. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-027, Revision 2
- F. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-043, Revision 2
- G. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-054, Revision 1
- H. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-056, Revision 1
- I. Perry Nuclear Power Plant, 10 CFR 50.55a Request PT-001, Revision 2

cc: NRC Region III Administrator
NRC Resident Inspector
Nuclear Reactor Regulation Project Manager

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Notification of Impracticality
In Accordance with 10 CFR 50.55a(g)(5)(iii)

--Inservice Inspection Impracticality--

1. **American Society of Mechanical Engineers (ASME) Code Components Affected**

Reactor Pressure Vessel (RPV) Meridional Head Welds	Code Class
1-B13-DG, Bottom Head Center Plate to Side Plates, 270° Side	1
1-B13-DH, Bottom Head Center Plate to Side Plates, 90° Side	1

2. **Applicable Code Edition and Addenda**

ASME Section XI, 2001 Edition through the 2003 Addenda

3. **Applicable Code Requirement**

Table IWB-2500-1, Examination Category B-A, Item No. B1.22 requires that essentially 100 percent of the RPV bottom head meridional weld examination volume, as defined by Figure IWB-2500-3, be examined.

4. **Impracticality of Compliance**

Examining essentially 100 percent of the weld volume for the affected components, as required by the ASME Code, is impractical. Weld volume examination limitations exist due to RPV bottom head interferences inherent to the BWR6 reactor design. Table IR-001-1 lists weld lengths available for volumetric examination; the accompanying sketch identifies existing, as-fabricated RPV bottom head interferences.

5. **Burden Caused by Compliance**

The bottom of the RPV was designed and fabricated such that the control rod drive (CRD) housings obstruct large portions of the bottom head meridional welds, causing a limitation in scanning coverage for the welds in this request. The RPV skirt support knuckle also impacts scanning coverage. Removing the obstructions would require the RPV bottom head to be redesigned and modified, which is impractical.

6. **Alternative and Basis for Use**

In lieu of examining essentially 100 percent of the RPV bottom head meridional weld examination volume, FENOC would examine 29 percent, which represents 80 inches of the 274-inch long weld.

Examinations meeting the requirements of ASME Section XI, except complete coverage for the entire length of the welds, would continue to be performed on the accessible lengths of the welds. Although the entire length of the weld cannot be

examined, the required volume of the accessible lengths would receive full coverage. Since the construction, operating conditions, and environmental conditions of the unexamined portions are identical to the examined portions, it is reasonable to apply the satisfactory results from the examined portions to the unexamined portions.

In accordance with Table IWB-2500-1, Examination Category B-P, VT-2 examinations of the reactor's pressure retaining components are performed during the RPV system leakage test prior to plant startup following a refueling outage. During this test, the RPV bottom head area and portions of the subject welds are physically accessible, and the physically accessible welds can be examined directly.

The structural integrity of the RPV welds was demonstrated during construction by meeting the requirements of ASME Code Section III, and by the requirements of ASME Section XI during preservice and subsequent inspections. No reportable indications were identified during the preservice, first, and second interval inspections.

Catastrophic reactor vessel failure is precluded by avoiding nil ductile temperatures at significant stress levels, where failure is brittle. None of the welds in this request are located in the beltline region of the reactor vessel, which is the region where neutron fluence can cause materials to become more brittle over time.

In conclusion, because the RPV has been free of reportable indications, the capability to perform partial ultrasonic (volume) examinations and the ability to perform direct VT-2 examinations of the physically accessible RPV bottom head area provide reasonable assurance that the structural integrity of the subject welds is acceptable.

7. Duration of Alternative

This alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

8. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, November 22, 1999, Subject: Safety Evaluation of the Inservice Inspection Program Second 10-Year Interval Requests for Relief for FirstEnergy Nuclear Operating Company – Perry Nuclear Power Plant, Unit 1 (TAC No. MA3437).

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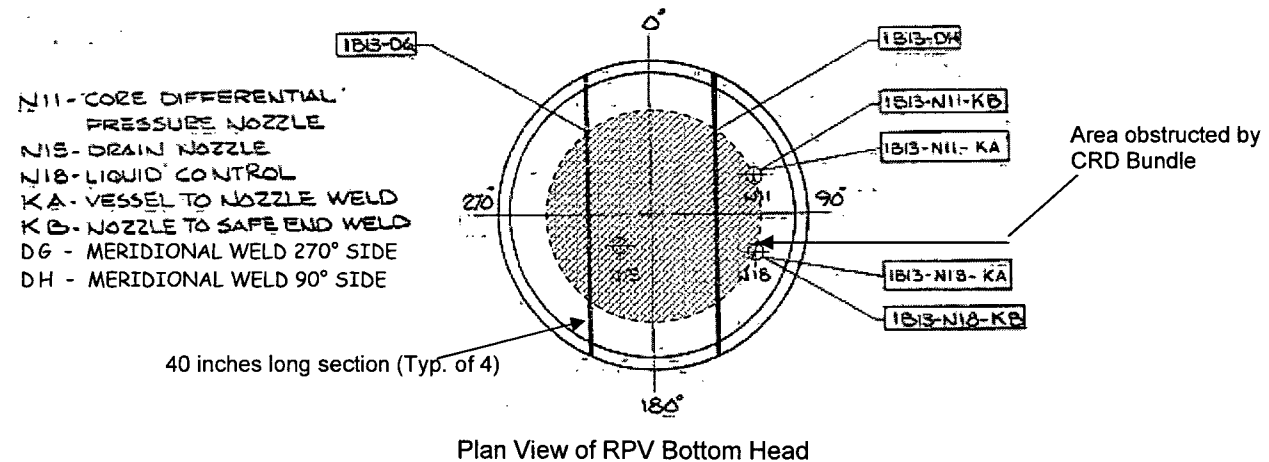
TABLE IR-001-1

SYSTEM: Reactor Pressure Vessel (B13)

ITEM NO.	WELD	DESCRIPTION	CODE CATEGORY	PERCENT COMPLETE		
				⊥	//	
B1.22	1-B13-DG	BOTTOM HEAD CENTER PLATE TO SIDE PLATES, 270° SIDE	B-A	29*	29*	OBSTRUCTION PRESENTED BY CRD HOUSING BUNDLE AND SKIRT SUPPORT KNUCKLE.
B1.22	1-B13-DH	BOTTOM HEAD CENTER PLATE TO SIDE PLATES, 90° SIDE	B-A	29*	29*	OBSTRUCTION PRESENTED BY CRD HOUSING BUNDLE AND SKIRT SUPPORT KNUCKLE.

⊥ PERPENDICULAR SCAN
// PARALLEL SCAN

* 29 percent represents completion of 80 inches of the 274-inch long weld by examining a 40-inch long section of the weld between the CRD bundle and vessel skirt support knuckle at each end of the weld (see sketch below)



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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. American Society of Mechanical Engineers (ASME) Code Components Affected

Reactor Pressure Vessel (RPV) Control Rod Drive (CRD) Housing Welds	Code Class
1-B13-02/23-FW, CRD Housing to Flange	1
1-B13-02/27-FW, CRD Housing to Flange	1
1-B13-02/31-FW, CRD Housing to Flange	1
1-B13-02/35-FW, CRD Housing to Flange	1
1-B13-02/39-FW, CRD Housing to Flange	1
1-B13-06/15-FW, CRD Housing to Flange	1
1-B13-06/47-FW, CRD Housing to Flange	1
1-B13-10/11-FW, CRD Housing to Flange	1
1-B13-10/51-FW, CRD Housing to Flange	1
1-B13-14/07-FW, CRD Housing to Flange	1
1-B13-14/55-FW, CRD Housing to Flange	1
1-B13-22/03-FW, CRD Housing to Flange	1
1-B13-22/59-FW, CRD Housing to Flange	1
1-B13-26/03-FW, CRD Housing to Flange	1
1-B13-26/59-FW, CRD Housing to Flange	1
1-B13-30/03-FW, CRD Housing to Flange	1
1-B13-30/59-FW, CRD Housing to Flange	1
1-B13-34/03-FW, CRD Housing to Flange	1
1-B13-34/59-FW, CRD Housing to Flange	1
1-B13-38/03-FW, CRD Housing to Flange	1
1-B13-38/59-FW, CRD Housing to Flange	1
1-B13-46/07-FW, CRD Housing to Flange	1
1-B13-46/55-FW, CRD Housing to Flange	1
1-B13-50/11-FW, CRD Housing to Flange	1
1-B13-50/51-FW, CRD Housing to Flange	1
1-B13-54/15-FW, CRD Housing to Flange	1
1-B13-54/47-FW, CRD Housing to Flange	1
1-B13-58/23-FW, CRD Housing to Flange	1
1-B13-58/27-FW, CRD Housing to Flange	1
1-B13-58/31-FW, CRD Housing to Flange	1
1-B13-58/35-FW, CRD Housing to Flange	1
1-B13-58/39-FW, CRD Housing to Flange	1

2. Applicable Code Edition and Addenda

ASME Section XI, 2001 Edition through the 2003 Addenda

3. Applicable Code Requirement

Table IWB-2500-1, Examination Category B-O, Item No. B14.10 requires essentially 100 percent of the RPV CRD housing weld surface or volumetric examination volume, as defined by Figure IWB-2500-18, for 10 percent of the peripheral CRD housings, to be examined.

4. Reason for Request

Insert and withdraw control line interferences create partial inaccessibility of the RPV CRD housing to flange welds. As depicted in Figure IR-009-1, the control lines limit examination coverage to 85 percent. Normal disassembly of the CRD mechanisms does not facilitate additional coverage for the required surface examination areas.

5. Proposed Alternative and Basis for Use

In lieu of examining essentially 100 percent of the RPV CRD housing weld surface or volumetric examination volume, FENOC proposes to examine the accessible portions (85 percent) of the identified welds.

Examinations that meet the requirements of ASME Section XI would be performed on 4 of the 32 peripheral CRD housing to flange welds. In rounding up to 4 welds (10 percent of 32 equals 3.2, which is rounded up to 4), the total weld length of the CRD housing welds examined exceeds 10 percent of the total length of all 32 of the peripheral CRD housing welds.

Unexamined portions of welds are subject to the same operating and environmental conditions as the examined portions. Therefore, it is reasonable to apply the results from examined weld portions to the unexamined portions.

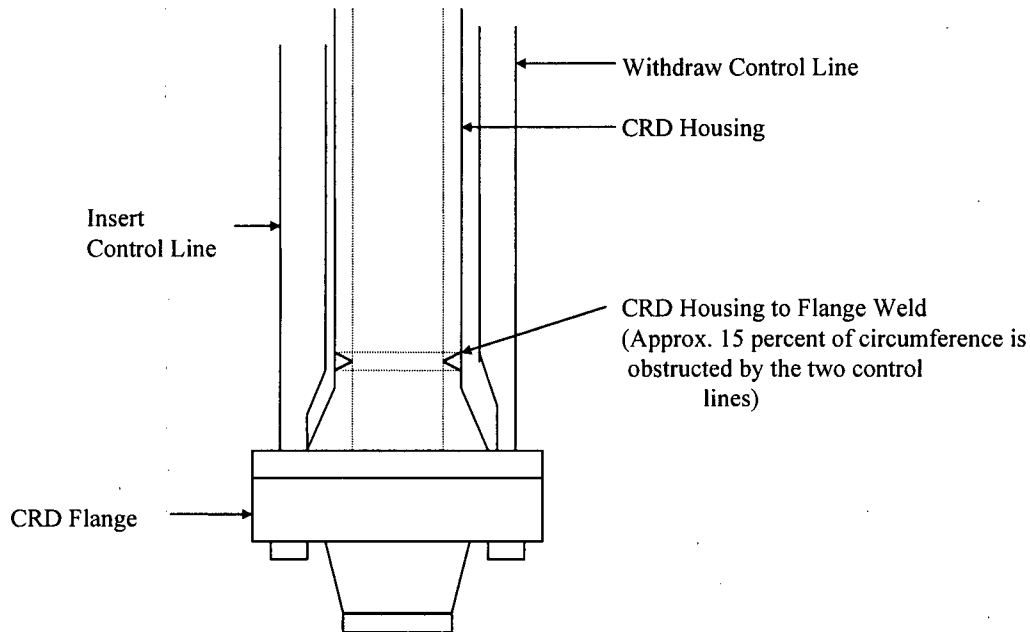


Figure IR-009-1

In summary, because of the capability to perform partial examinations (85 percent) of each of the selected welds, and examining more than 10 percent of the total length of all of the peripheral CRD housing to flange welds, it is concluded that the proposed alternative provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, November 22, 1999, Subject: Safety Evaluation of the Inservice Inspection Program Second 10-Year Interval Requests for Relief for FirstEnergy Nuclear Operating Company – Perry Nuclear Power Plant, Unit 1 (TAC No. MA3437).

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)

--Hardship or Unusual Difficulty
without Compensating Increase in Level of Quality or Safety--

1. **American Society of Mechanical Engineers (ASME) Code Components Affected**

Reactor Core Isolation Cooling (RCIC) System Pump Support Welded Attachment	Code Class
1-E51-C0001-A-WA, Welded Pump Casing Support Lug	2

2. **Applicable Code Edition and Addenda**

ASME Section XI, 2001 Edition through the 2003 Addenda

3. **Applicable Code Requirement**

Table IWC-2500-1, Examination Category C-C, Item No. C3.30 requires 100 percent of the required areas of the welded attachment surface, as defined by Figure IWC-2500-5, to be examined. Note 5 of the table allows for a sample of 10 percent of the welded attachments.

4. **Reason for Request**

The pump support integral attachment weld is not fully accessible due to pedestal interference. As depicted in Figure IR-012-1, coverage is limited to approximately 83 percent. Disassembly of the pump pedestal to provide additional coverage does not significantly improve the quality of the examination without an increase in hardship from the maintenance (rework/reassembly) or redesign of the pedestal.

5. **Proposed Alternative and Basis for Use**

In lieu of examining 100 percent of required areas of the welded attachment surface, FENOC proposes to examine 83 percent.

An examination that meets the requirements of ASME Section XI, except coverage, would be performed.

Since the construction, operating conditions and environmental conditions of the unexamined portions of the weld are identical to the examined portions, it is reasonable to apply satisfactory results to the unexamined portions.

Full access to the subject pump casing integral attachment welds could only be provided by disassembly or redesign of the pump's pedestal mounting.

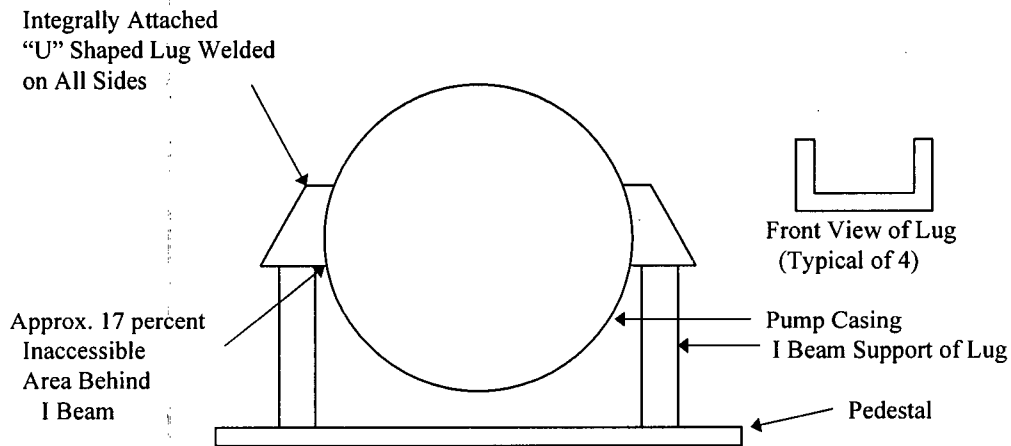


Figure IR-012-1

In summary, because of the capability to examine at least 83 percent of the weld surfaces on a continuing basis, it is concluded that performing the applicable Code examinations would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, November 22, 1999, Subject: Safety Evaluation of the Inservice Inspection Program Second 10-Year Interval Requests for Relief for FirstEnergy Nuclear Operating Company – Perry Nuclear Power Plant, Unit 1 (TAC No. MA3437).

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)

--Hardship or Unusual Difficulty
without Compensating Increase in Level of Quality or Safety--

1. **American Society of Mechanical Engineers (ASME) Code Components Affected**

Pump Casing Welds	Code Class
1-E22-C001-001, High Pressure Core Spray Pump, Head to Barrel Shell	2
1-E22-C001-002, High Pressure Core Spray Pump, Shell to Shell	2
1-E22-C001-003, High Pressure Core Spray Pump, Shell to Shell	2
1-E22-C001-004, High Pressure Core Spray Pump, Shell to Flange	2
1-E22-C001-013, High Pressure Core Spray Pump, Barrel Longseam	2
1-E22-C001-014, High Pressure Core Spray Pump, Barrel Longseam	2
1-E22-C001-015, High Pressure Core Spray Pump, Barrel Longseam	2
1-E21-C001-001, Low Pressure Core Spray Pump, Head to Barrel Shell	2
1-E21-C001-002, Low Pressure Core Spray Pump, Shell to Shell	2
1-E21-C001-003, Low Pressure Core Spray Pump, Shell to Flange	2
1-E21-C001-012, Low Pressure Core Spray Pump, Shell to Shell	2
1-E21-C001-013, Low Pressure Core Spray Pump, Barrel Longseam	2
1-E21-C001-014, Low Pressure Core Spray Pump, Barrel Longseam	2
1-E21-C001-015, Low Pressure Core Spray Pump, Barrel Longseam	2
1-E12-C002A-001, Residual Heat Removal Pump A, Head to Barrel Shell	2
1-E12-C002A-002, Residual Heat Removal Pump A, Shell to Shell	2
1-E12-C002A-003, Residual Heat Removal Pump A, Shell to Flange	2
1-E12-C002A-012, Residual Heat Removal Pump A, Shell to Shell	2
1-E12-C002A-013, Residual Heat Removal Pump A, Barrel Longseam	2
1-E12-C002A-014, Residual Heat Removal Pump A, Barrel Longseam	2
1-E12-C002A-015, Residual Heat Removal Pump A, Barrel Longseam	2
1-E12-C002B-001, Residual Heat Removal Pump B, Head to Barrel Shell	2
1-E12-C002B-002, Residual Heat Removal Pump B, Shell to Shell	2
1-E12-C002B-003, Residual Heat Removal Pump B, Shell to Flange	2
1-E12-C002B-012, Residual Heat Removal Pump B, Shell to Shell	2
1-E12-C002B-013, Residual Heat Removal Pump B, Barrel Longseam	2
1-E12-C002B-014, Residual Heat Removal Pump B, Barrel Longseam	2
1-E12-C002B-015, Residual Heat Removal Pump B, Barrel Longseam	2
1-E12-C002C-001, Residual Heat Removal Pump C, Head to Barrel Shell	2
1-E12-C002C-002, Residual Heat Removal Pump C, Shell to Shell	2
1-E12-C002C-003, Residual Heat Removal Pump C, Shell to Flange	2
1-E12-C002C-012, Residual Heat Removal Pump C, Shell to Shell	2
1-E12-C002C-013, Residual Heat Removal Pump C, Barrel Longseam	2
1-E12-C002C-014, Residual Heat Removal Pump C, Barrel Longseam	2
1-E12-C002C-015, Residual Heat Removal Pump C, Barrel Longseam	2

2. **Applicable Code Edition and Addenda**

ASME Section XI, 2001 Edition through the 2003 Addenda

3. Applicable Code Requirement

Table IWC-2500-1, Examination Category C-G, Item No. C6.10 requires surface examination of the areas defined by Figure IWC-2500-8 to be examined. Note 1 of the table states that in the case of multiple pumps of similar design, size, function and service in a system, required weld examinations may be limited to all the welds in one pump in the same group.

4. Reason for Request

The pump casing welds are inaccessible because they are located within those portions of vertical line shaft pump casings that are installed below the floor.

5. Proposed Alternative and Basis for Use

In accordance with the In-Service Inspection Plan, if the subject welds become accessible through the disassembly of the pumps for maintenance, repair or modification, examinations that meet the requirements of ASME Section XI shall be performed. Only accessible welds are scheduled for examination at this time.

Figure IR-013-1 shows an example of the inaccessible welds in Residual Heat Removal Pump A; however, the other pumps listed in this request are similar. As depicted in Figure IR-013-1, the subject welds are below the floor elevation. Only disassembly of the pump and removal of the pump internals would provide examination access to these welds. For example, disassembly of the Residual Heat Removal Pump A would involve removal of the pump motor (approximately 7,800 pounds), and the pump head and vertical line shaft assembly (approximately 22 feet long and 16,000 pounds).

The pump casing welds above the floor elevation are fully accessible for examination. Since the construction and operating conditions of these pump casing welds are identical to those of the inaccessible welds, it is reasonable to apply satisfactory results from examined welds to the unexamined welds.

In summary, because of their acceptable initial condition and the capability to examine the similar accessible welds on a continuing basis, it is concluded that performing the applicable Code examinations would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. **Precedent**

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, November 22, 1999, Subject: Safety Evaluation of the Inservice Inspection Program Second 10-Year Interval Requests for Relief for FirstEnergy Nuclear Operating Company – Perry Nuclear Power Plant, Unit 1 (TAC No. MA3437).

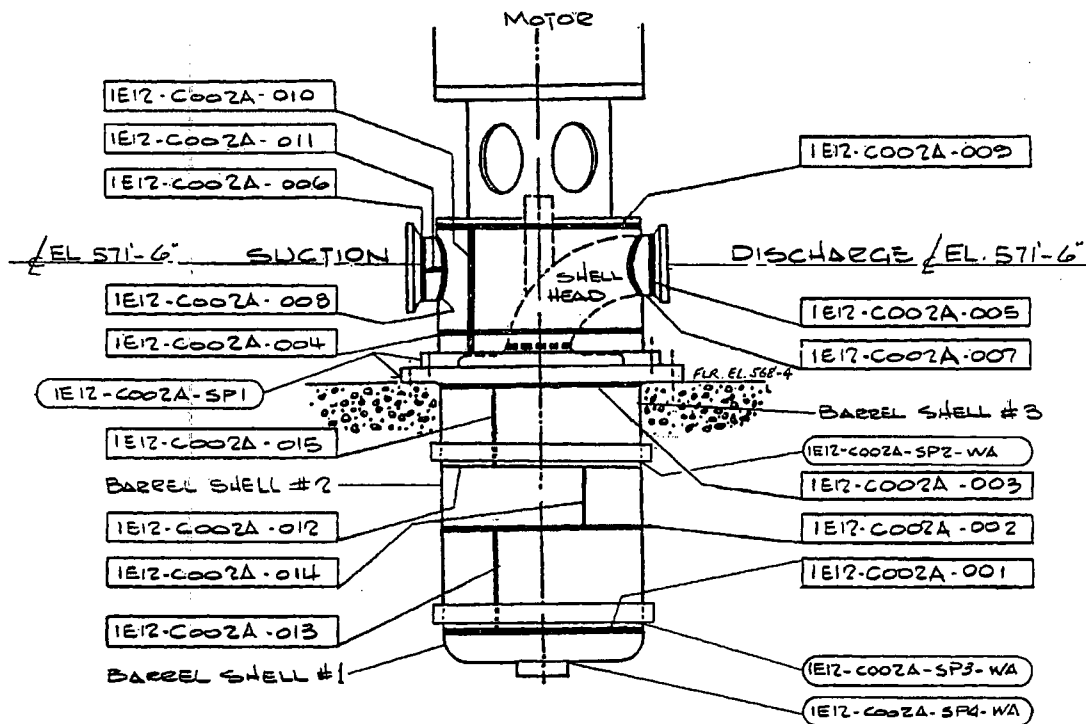


Figure IR-013-1

Note: This figure shows an example (Residual Heat Removal Pump A – 1-E12) of the inaccessible welds; other pumps in this request are similar.

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)

--Hardship or Unusual Difficulty
without Compensating Increase in Level of Quality or Safety--

1. **American Society of Mechanical Engineers (ASME) Code Components Affected**

Pressure Vessel Welded Attachment	Code Class
1R45-A003A-WA, Standby Diesel Generator Integrally Attached Anchor of Division 1 Diesel Fuel Oil Day Tank	3
1R45-A003B-WA, Standby Diesel Generator Integrally Attached Anchor of Division 2 Diesel Fuel Oil Day Tank	3

2. **Applicable Code Edition and Addenda**

ASME Section XI, 2001 Edition through the 2003 Addenda

3. **Applicable Code Requirement**

Table IWD-2500-1, Examination Category D-A, Item No. D1.10 requires visual examination (VT-1) of 100 percent of the required areas of each pressure vessel welded attachment. Note 3 of the table states for multiple vessels of similar design, function, and service, the welded attachments of only one of the multiple vessels shall be selected for examination.

4. **Reason for Request**

Access limitations due to the fire retardant coating (Pyrocrete) on the welded attachment make it difficult to perform VT-1 examination of the attachment.

5. **Proposed Alternative and Basis for Use**

At the time of the scheduled ASME Code Section XI, Examination Category F-A visual examinations of the day tank anchor, the Pyrocrete covering the integral attachment would be examined for any condition that might indicate that the integral attachments are structurally degraded (examples include, severely cracked or missing Pyrocrete and support detached from component). The first and second 10-year interval examinations produced acceptable results with no visible signs of structural degradation.

Pursuant to the Perry Nuclear Power Plant (PNPP) Fire Protection Program requirements (based on 10 CFR 50 Appendix R and Branch Technical position APCSB 9.5-1, Appendix A), the integrally attached (welded) anchor on the fuel oil day tank is buried in fire retardant Pyrocrete. Pyrocrete is a hard, rigid material. When

applied, it is considered a permanent feature of the system to endure through the life span of the facility. To remove this material from the day tank would require cutting and chipping.

The structural integrity of the pressure boundary was demonstrated during construction, prior to application of Pyrocrete, by meeting the requirements of ASME Section III.

In summary, because of its acceptable initial condition and the capability to visually examine the Pyrocrete for indications of degradation of the underlying attachment welds, it is concluded that performing the applicable Code examination would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, November 22, 1999, Subject: Safety Evaluation of the Inservice Inspection Program Second 10-Year Interval Requests for Relief for FirstEnergy Nuclear Operating Company – Perry Nuclear Power Plant, Unit 1 (TAC No. MA3437).

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)

--Hardship or Unusual Difficulty
without Compensating Increase in Level of Quality and Safety--

1. **American Society of Mechanical Engineers (ASME) Code Components Affected**

Valve Body Welds	Code Class
1G33-F0101-SEAM, Reactor Water Clean-Up, 3" Gate Valve	1
1G33-F0100-SEAM, Reactor Water Clean-Up, 4" Gate Valve	1
1G33-F0106-SEAM, Reactor Water Clean-Up, 4" Gate Valve	1
1G33-F0001-SEAM, Reactor Water Clean-Up, 6" Gate Valve	1
1G33-F0004-SEAM, Reactor Water Clean-Up, 6" Gate Valve	1
1E12-F0019-SEAM, Residual Heat Removal, 6" Check Valve	1
1E51-F0013-SEAM, Reactor Core Isolation Cooling, 6" Gate Valve	1
1E51-F0063-SEAM, Reactor Core Isolation Cooling, 10" Gate Valve	1
1E51-F0064-SEAM, Reactor Core Isolation Cooling, 10" Gate Valve	1
1E12-F0039A-SEAM, Residual Heat Removal, 12" Gate Valve	1
1E12-F0039B-SEAM, Residual Heat Removal, 12" Gate Valve	1
1E12-F0039C-SEAM, Residual Heat Removal, 12" Gate Valve	1
1E12-F0042A-SEAM, Residual Heat Removal, 12" Gate Valve	1
1E12-F0042B-SEAM, Residual Heat Removal, 12" Gate Valve	1
1E12-F0042C-SEAM, Residual Heat Removal, 12" Gate Valve	1
1E21-F0005-SEAM, Low Pressure Core Spray, 12" Gate Valve	1
1E21-F0007-SEAM, Low Pressure Core Spray, 12" Gate Valve	1
1E22-F0036-SEAM, High Pressure Core Spray, 12" Gate Valve	1

2. **Applicable Code Edition and Addenda**

ASME Section XI, 2001 Edition through the 2003 Addenda

3. **Applicable Code Requirement**

Table IWB-2500-1, Examination Category B-M-1 requires surface examination of valve body pressure retaining welds in valves less than 4-inch nominal pipe size (NPS) (Item No. B12.30) and volumetric examination of valve body pressure retaining welds in valves 4-inch NPS and larger (Item No. B12.40) as defined by Figure IWB-2500-17. Note 3 of the table states that the examinations are limited to at least one valve within each group of valves that are of the same size, construction design (such as globe, gate, or check valves), and manufacturing method, and that perform similar functions in the system, such as containment isolation or overpressure protection.

4. Reason for Request

Performing surface and volumetric examinations on Category B-M-1 pressure retaining welds in valve bodies results in unnecessary occupational radiation exposure to nondestructive examination (NDE) personnel and support workers, such as insulators and scaffold builders.

5. Proposed Alternative and Basis for Use

In accordance with Examination Category B-P, the welds receive a VT-2 examination each refueling outage during the performance of the system leakage test of the Class 1 boundary.

The structural integrity of the pressure boundary was demonstrated during construction by meeting the requirements of ASME Section III and ASME Section XI during preservice and in-service examinations with no relevant indications identified.

A search of industry operating experience did not identify any failures of valve body welds. As a result of their excellent performance, the 2008 addenda to ASME Section XI deleted Category B-M-1 valve body weld examinations. Risk-informed insights have not identified any degradation mechanism specifically associated with these welds. These examinations result in unnecessary radiation exposure to NDE and support personnel. Degradation of the valve interior would be detected by the Category B-L-2 and B-M-2 [visual testing] VT-1 examinations or by the mechanic working on the component internals, and through-wall leakage would be detected by the VT-2 examinations during system pressure tests.

As required by Table IWB-2500-1, Note 3, only 8 of the 18 identified valve body welds require examination (one valve in each of the groups). Dose surveys show dose rates at the subject valves as high as 2,500 mRem/hour. Approximately one hour must be spent at each valve location to perform insulation removal and reinstallation, and the examination. It is estimated that eliminating the Category B-M-1 required examinations for these eight valve body welds would provide a collective dose savings of at least 4,600 mRem.

In summary, due to satisfactory valve body weld performance, absence of a degradation mechanism, the deletion of Category B-M-1 examinations in the ASME Code's 2008 Addenda, and the ability to detect through-wall leakage during VT-2 system pressure tests, it is concluded that the proposed alternative provides an acceptable level of quality and safety while eliminating unnecessary radiation exposure to NDE personnel and support workers.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, August 22, 2008, Subject: Perry Nuclear Power Plant, Unit No. 1 – Request to Eliminate Selected Volumetric Examinations of Cast Valve Body Welds for the Second 10-Year Inservice Inspection Interval (TAC No. MD8196).

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. American Society of Mechanical Engineers (ASME) Code Components Affected

Reactor Pressure Vessel Welds	Code Class
1B13-N1A-KA, 22" Recirculation Outlet Nozzle N1A to Vessel	1
1B13-N1A-IR, 22" Recirculation Outlet Nozzle N1A Inner Radius	1
1B13-N1B-KA, 22" Recirculation Outlet Nozzle N1B to Vessel	1
1B13-N1B-IR, 22" Recirculation Outlet Nozzle N1B Inner Radius	1
1B13-N2A-KA, 12" Recirculation Inlet Nozzle N2A to Vessel	1
1B13-N2A-IR, 12" Recirculation Inlet Nozzle N2A Inner Radius	1
1B13-N2B-KA, 12" Recirculation Inlet Nozzle N2B to Vessel	1
1B13-N2B-IR, 12" Recirculation Inlet Nozzle N2B Inner Radius	1
1B13-N2C-KA, 12" Recirculation Inlet Nozzle N2C to Vessel	1
1B13-N2C-IR, 12" Recirculation Inlet Nozzle N2C Inner Radius	1
1B13-N2D-KA, 12" Recirculation Inlet Nozzle N2D to Vessel	1
1B13-N2D-IR, 12" Recirculation Inlet Nozzle N2D Inner Radius	1
1B13-N2E-KA, 12" Recirculation Inlet Nozzle N2E to Vessel	1
1B13-N2E-IR, 12" Recirculation Inlet Nozzle N2E Inner Radius	1
1B13-N2F-KA, 12" Recirculation Inlet Nozzle N2F to Vessel	1
1B13-N2F-IR, 12" Recirculation Inlet Nozzle N2F Inner Radius	1
1B13-N2G-KA, 12" Recirculation Inlet Nozzle N2G to Vessel	1
1B13-N2G-IR, 12" Recirculation Inlet Nozzle N2G Inner Radius	1
1B13-N2H-KA, 12" Recirculation Inlet Nozzle N2H to Vessel	1
1B13-N2H-IR, 12" Recirculation Inlet Nozzle N2H Inner Radius	1
1B13-N2J-KA, 12" Recirculation Inlet Nozzle N2J to Vessel	1
1B13-N2J-IR, 12" Recirculation Inlet Nozzle N2J Inner Radius	1
1B13-N2K-KA, 12" Recirculation Inlet Nozzle N2K to Vessel	1
1B13-N2K-IR, 12" Recirculation Inlet Nozzle N2K Inner Radius	1
1B13-N3A-KA, 26" Main Steam Nozzle N3A to Vessel	1
1B13-N3A-IR, 26" Main Steam Nozzle N3A Inner Radius	1
1B13-N3B-KA, 26" Main Steam Nozzle N3B to Vessel	1
1B13-N3B-IR, 26" Main Steam Nozzle N3B Inner Radius	1
1B13-N3C-KA, 26" Main Steam Nozzle N3C to Vessel	1
1B13-N3C-IR, 26" Main Steam Nozzle N3C Inner Radius	1
1B13-N3D-KA, 26" Main Steam Nozzle N3D to Vessel	1
1B13-N3D-IR, 26" Main Steam Nozzle N3D Inner Radius	1
1B13-N5A-KA, 12" Core Spray Nozzle N5A to Vessel	1
1B13-N5A-IR, 12" Core Spray Nozzle N5A Inner Radius	1
1B13-N5B-KA, 12" Core Spray Nozzle N5B to Vessel	1
1B13-N5B-IR, 12" Core Spray Nozzle N5B Inner Radius	1
1B13-N6A-KA, 12" Low Pressure Core Injection N6A to Vessel	1

Reactor Pressure Vessel Welds	Code Class
1B13-N6A-IR, 12" Low Pressure Core Injection N6A Inner Radius	1
1B13-N6B-KA, 12" Low Pressure Core Injection N6B to Vessel	1
1B13-N6B-IR, 12" Low Pressure Core Injection N6B Inner Radius	1
1B13-N6C-KA, 12" Low Pressure Core Injection N6C to Vessel	1
1B13-N6C-IR, 12" Low Pressure Core Injection N6C Inner Radius	1
1B13-N7-KA, 6" Top Head Spray Spare Nozzle N7 to Vessel	1
1B13-N7-IR, 6" Top Head Spray Spare Nozzle N7 Inner Radius	1
1B13-N8-KA, 6" Top Head Spray Nozzle N8 to Vessel	1
1B13-N8-IR, 6" Top Head Spray Nozzle N8 Inner Radius	1
1B13-N9A-KA, 4" Jet Pump Instrumentation Nozzle N9A to Vessel	1
1B13-N9A-IR, 4" Jet Pump Instrumentation Nozzle N9A Inner Radius	1
1B13-N9B-KA, 4" Jet Pump Instrumentation Nozzle N9B to Vessel	1
1B13-N9B-IR, 4" Jet Pump Instrumentation Nozzle N9B Inner Radius	1

2. Applicable Code Edition and Addenda

ASME Section XI, 2001 Edition through the 2003 Addenda

3. Applicable Code Requirement

Table IWB-2500-1, Examination Category B-D, requires volumetric examination of full penetration nozzle-to-vessel (Item No. B3.90) and nozzle inside radius section (Item No. B3.100) welds, as defined by Figures IWB-2500-7(a) through (d) for 100 percent of the total population each interval.

4. Reason for Request

Without approval to incorporate Code Case N-702, all Class 1 nozzle-to-vessel welds and nozzle inner radii section welds require examination during the third in-service inspection interval.

5. Proposed Alternative and Basis for Use

In lieu of performing examination on 100 percent of the identified nozzle assemblies, FENOC proposes to perform, in accordance with Code Case N-702, examinations on a minimum of 25 percent of the nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size. For each of the identified nozzle assemblies, both the inner radius and the nozzle-to-shell weld would be examined. The following nozzle assemblies would be selected for examination: one of two 22-inch recirculation outlet nozzle assemblies; three of the ten 12-inch recirculation inlet nozzle assemblies, one of the four 26-inch main steam nozzle assemblies; one of the two 12-inch core spray nozzle assemblies; one of the three 12-inch low pressure core injection nozzle assemblies, one of the two 6-inch head spray nozzle assemblies, and one of the two 4-inch jet pump instrumentation nozzle assemblies.

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Code Case N-702 proposes that VT-1 visual examination may be used in lieu of volumetric examination for the inner radii (Item B3.100). The Perry Nuclear Power Plant (PNPP) is already using Code Case N-648-1 in accordance with conditions placed upon the use of Code Case N-648-1 by Regulatory Guide 1.147, which allows VT-1 visual examination for nozzle inner radii. As Code Case N-648-1 is already approved for use at the PNPP, the specific aspect of utilizing VT-1 visual examinations as allowed by Code Case N-702 is not a part of the request. Despite this allowance, volumetric examinations of the nozzle inner radii of the selected recirculation inlet, core spray, low pressure core injection, and jet pump instrumentation nozzles are performed as their nozzle inner radii are not fully accessible from inside the vessel.

EPRI Technical Report 1003557 (Reference 1) provides the basis for Code Case N-702. The evaluation found that failure probabilities due to a low temperature overpressure event at the nozzle blend radius region and nozzle-to-vessel shell weld are very low (that is, $< 1 \times 10^{-6}$ for 40 years) with or without in-service inspection. The report concludes that inspection of 25 percent of each nozzle type is technically justified.

On December 19, 2007, the Nuclear Regulatory Commission (NRC) issued a safety evaluation (SE) approving BWRVIP-108 as a basis for using Code Case N-702. Within Section 5 of the SE, it states that each licensee should demonstrate the plant-specific applicability of the BWRVIP-108 report to their units in the relief request by meeting the criteria discussed in Section 5 of the SE.

The applicability of the BWRVIP-108 report to the PNPP is demonstrated by showing the criteria within Section 5 of the SE are met.

The generic terms to be used in the SE Section 5 applicability evaluations are:

C_{RPV} = recirculation inlet or outlet nozzles (from BWRVIP-108 model)
 $C_{RPV} = 19332 \text{ psi}$ (recirculation inlet nozzles)
 $C_{RPV} = 16171 \text{ psi}$ (recirculation outlet nozzles)
 C_{NOZZLE} = recirculation inlet or outlet nozzles (from BWRVIP-108 model)
 $C_{NOZZLE} = 1637 \text{ psi}$ (recirculation inlet nozzles)
 $C_{NOZZLE} = 1977 \text{ psi}$ (recirculation outlet nozzles)

The PNPP-specific terms to be used in the SE Section 5 applicability evaluations are:

Heatup/Cooldown rate = $100^{\circ}\text{F}/\text{hour}$
 p = reactor pressure vessel (RPV) normal operating pressure, $p = 1045 \text{ psig}$
(maximum reactor steam dome pressure per Technical Specification 3.4.12)
 r = RPV inner radius, $r = 119"$
 t = RPV wall thickness, $t = 7.19"$
 r_i = nozzle inner radius
 r_i = inner radius for recirculation outlet (N1) nozzles, $r_i = 10"$

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r_i = inner radius for recirculation inlet (N2) nozzles, $r_i = 5.813"$
 r_o = nozzle outer radius
 r_o = outer radius for recirculation outlet (N1) nozzles, $r_o = 17.594"$
 r_o = outer radius for recirculation inlet (N2) nozzles, $r_o = 11.125"$

Given the generic and plant-specific terms, the PNPP conformance with the five criteria is demonstrated as follows:

(1) Max RPV Heatup/Cooldown Rate°F/hour

Criterion – the maximum RPV heatup/cooldown rate is limited to $< 115^\circ\text{F}/\text{hour}$

In accordance with Technical Specification 3.4.11, reactor coolant system heatup and cooldown rates are maintained at $\leq 100^\circ\text{F}$ in any one hour period.

By letter dated September 17, 2008 (Accession No. ML082680091), PNPP provided a response to a request for additional information (RAI) regarding plant operation data in recent years with respect to heatups and cooldowns exceeding 115°F per hour. As documented in the safety evaluation for Revision 0 of this request (Accession No. ML082960729), the NRC staff agreed with PNPP that events in which the 115°F heat-up/cool-down rate was exceeded were transients and as a result the first criterion regarding the maximum RPV heat-up/cool-down rate (which is intended to address normal operating conditions) was satisfied. Since 2008, no other events have occurred where normal heatup/cool-down rates exceeded $115^\circ\text{F}/\text{hour}$.

(2) Recirculation Inlet (N2) Nozzles

Equation to meet criterion: $(pr/t)/C_{RPV} < 1.15$

$$(1045 \times 119 \div 7.19) \div 19332 < 1.15$$

The PNPP result is 0.89, which is less than 1.15.

(3) Recirculation Inlet (N2) Nozzles

Equation to meet criterion: $[p(r_o^2 + r_i^2) / (r_o^2 - r_i^2)] / C_{NOZZLE} < 1.15$

$$[1045 \times (11.125^2 + 5.813^2) \div (11.125^2 - 5.813^2)] \div 1637 < 1.15$$

The PNPP result is 1.12, which is less than 1.15.

(4) Recirculation Outlet (N1) Nozzles

Equation to meet criterion: $(pr/t)/C_{RPV} < 1.15$

$$(1045 \times 119 \div 7.19) \div 16171 < 1.15$$

The PNPP result is 1.07, which is less than 1.15.

(5) Recirculation Outlet (N1) Nozzles

Equation to meet criterion: $[p(r_o^2 + r_i^2) / (r_o^2 - r_i^2)] / C_{NOZZLE} < 1.15$

$$[1045 \times (17.594^2 + 10^2) \div (17.594^2 - 10^2)] \div 1977 < 1.15$$

The PNPP result is 1.03, which is less than 1.15.

The results of the above equations demonstrate the applicability of the BWRVIP-108 report to the PNPP by showing the criteria within Section 5 of the NRC SE is met. Therefore, the basis for using Code Case N-702 is demonstrated for the PNPP.

Table 1 provides a synopsis of inspections already performed on the components for which this proposed alternative is requested, including disposition of any indications found.

The proposed alternative use of Code Case N-702 provides an acceptable level of quality and safety, and the reduction in scope is estimated to provide for a collective dose savings of as much as 15,000 mREM.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019, or until the Nuclear Regulatory Commission publishes Code Case N-702 in a future revision of Regulatory Guide 1.147.

7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, December 29, 2008, Subject: Perry Nuclear Power Plant, Unit No. 1 – Request for Relief Related to Inservice Inspection Relief Request IR-054 (TAC No. MD8458).

Nuclear Regulatory Commission letter to Exelon Nuclear, November 3, 2009, Subject: Dresden Nuclear Power Station, Units 2 and 3 – Alternative to Nozzle-To-Vessel Weld and Inner Radius Examinations (TAC Nos. ME0882 and ME0883).

Nuclear Regulatory Commission letter to Exelon Nuclear, August 24, 2009, Subject: Clinton Power Station, Unit No. 1 – Proposed Alternative to 10 CFR 50.55a Examination Requirements for Reactor Pressure Vessel Weld Inspections (TAC No. ME0218).

Nuclear Regulatory Commission letter to Exelon Nuclear, February 2, 2010, Subject: Quad Cities Nuclear Power Station, Unit Nos. 1 and 2 – Alternative to Nozzle to Vessel Weld and Inner Radius Examinations (TAC Nos. ME0765 and ME0766).

8. References

1. EPRI Technical Report 1003557, "BWRVIP-108: BWR Vessel and Internals Project Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," October 2002.
2. ASME Boiler and Pressure Vessel Code, Code Case N-648-1, "Alternative Requirements for Inner Radius Examinations of Class I Reactor Vessel Nozzles, Section XI, Division 1," September 7, 2001.
3. ASME Boiler and Pressure Vessel Code, Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1," February 20, 2004.
4. Matthew A. Mitchell, Office of Nuclear Reactor Regulation, to Rick Libra, BWRVIP Chairman, "Safety Evaluation of Proprietary EPRI Report, 'BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)'," December 19, 2007 (Accession No. ML073600374).

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Table 1

<u>Component</u>	<u>Outage(s) Inspected - Results*</u>
1B13-N1A-KA	1R5 – NRI 1R11 – NRI
1B13-N1A-IR	1R5 – NRI 1R11 – NRI
1B13-N1B-KA	1R5 – NRI
1B13-N1B-IR	1R5 – NRI
1B13-N2A-KA	1R5 – NRI
1B13-N2A-IR	1R5 – NRI
1B13-N2B-KA	1R5 – NRI 1R11 – One subsurface indication; acceptable.
1B13-N2B-IR	1R5 – NRI 1R11 – NRI
1B13-N2C-KA	1R2 – NRI
1B13-N2C-IR	1R2 – NRI
1B13-N2D-KA	1R5 – NRI
1B13-N2D-IR	1R5 – NRI
1B13-N2E-KA	1R5 – NRI 1R11 – NRI
1B13-N2E-IR	1R5 – NRI 1R11 – NRI
1B13-N2F-KA	1R5 – NRI
1B13-N2F-IR	1R5 – NRI
1B13-N2G-KA	1R5 – NRI
1B13-N2G-IR	1R5 – NRI
1B13-N2H-KA	1R5 – NRI
1B13-N2H-IR	1R5 – NRI
1B13-N2J-KA	1R2 – NRI
1B13-N2J-IR	1R2 – NRI
1B13-N2K-KA	1R5 – NRI 1R11 – NRI
1B13-N2K-IR	1R5 – NRI 1R11 – NRI
1B13-N3A-KA	1R1 – NRI 1R10 – NRI
1B13-N3A-IR	1R1 – NRI 1R10 – NRI
1B13-N3B-KA	1R5 – NRI 1R10 – NRI
1B13-N3B-IR	1R5 – NRI 1R10 – NRI
1B13-N3C-KA	1R5 – NRI 1R10 – NRI
1B13-N3C-IR	1R5 – NRI 1R10 – NRI

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Table 1 (continued)

<u>Component</u>	<u>Outage(s) Inspected – Results*</u>
1B13-N3D-KA	1R5 – NRI 1R10 – NRI
1B13-N3D-IR	1R5 – NRI 1R10 – NRI
1B13-N5A-KA	1R5 – NRI 1R8 – One subsurface indication; acceptable.
1B13-N5A-IR	1R5 – NRI 1R8 – NRI
1B13-N5B-KA	1R5 – NRI 1R7 – NRI
1B13-N5B-IR	1R5 – NRI 1R7 – NRI
1B13-N6A-KA	1R5 – NRI 1R8 – Four subsurface indications; acceptable.
1B13-N6A-IR	1R5 – NRI 1R8 – NRI
1B13-N6B-KA	1R5 – NRI 1R8 – NRI
1B13-N6B-IR	1R5 – NRI 1R8 – NRI
1B13-N6C-KA	1R2 – NRI 1R8 – NRI
1B13-N6C-IR	1R2 – NRI 1R8 – NRI
1B13-N7-KA	1R6 – NRI
1B13-N7-IR	1R6 – NRI
1B13-N8-KA	1R1 – NRI 1R8 – NRI
1B13-N8-IR	1R1 – NRI 1R8 – NRI
1B13-N9A-KA	1R2 – NRI
1B13-N9A-IR	1R2 – NRI
1B13-N9B-KA	1R5 – NRI 1R11 – NRI
1B13-N9B-IR	1R5 – NRI 1R11 – NRI

*NRI = No relevant indications; that is, no indications that required evaluation against ASME Section XI acceptance criteria.

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. **American Society of Mechanical Engineers (ASME) Code Components Affected**

Core Support Structure Components	Code Class
Reactor Vessel Interior	1
Shroud Support Plate	1
Shroud Support Legs	1
Shroud Horizontal Welds	1
Shroud Vertical Welds	1
Shroud Repairs	1
Top Guide and Top Guide Grid	1
Core Support Plate	1
Control Rod Guide Tubes	1

2. **Applicable Code Edition and Addenda**

ASME Section XI, 2001 Edition through the 2003 Addenda

3. **Applicable Code Requirement**

Table IWB-2500-1, Examination Category B-N-1, Item No. B13.10 requires accessible areas of the reactor vessel interior to be examined each inspection period by the visual, VT-3 method. Examination Category B-N-2, Item No. B13.40 requires accessible surfaces of the core support structure to the reactor vessel to be examined by the visual, VT-3 method each interval.

4. **Reason for Request**

FENOC requests to use the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines, endorsed by the Nuclear Regulatory Commission (NRC) and implemented by the industry, to perform examinations in accordance with industry initiatives because Code inspection requirements have not evolved with boiling water reactor (BWR) inspection experience.

5. **Proposed Alternative and Basis for Use**

In lieu of the ASME Section XI examination requirements, FENOC proposes to perform examinations pursuant to the requirements within the identified BWRVIP guidelines (NRC approved with the exception of BWRVIP-183, which is currently under NRC review).

The BWRVIP Inspection and Evaluation (I&E) guidelines have recommended aggressive specific inspection by Boiling Water Reactor (BWR) operators to identify material condition issues with BWR components. A wealth of inspection data has been gathered during these inspections across the BWR industry. The I&E guidelines focus on specific and susceptible components, specify appropriate inspection methods capable of identifying real anticipated degradation mechanisms, and require re-examination at conservative intervals. In contrast, the Code inspection requirements were prepared before the BWRVIP initiative and have not evolved with BWR inspection experience.

Not all the components addressed by these guidelines are Code components. The guidelines applicable to the subject Code components are:

BWRVIP-03, "Reactor Pressure Vessel and Internals Examination Guidelines"
BWRVIP-18-A, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"
BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines"
BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"
BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines"
BWRVIP-38, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines"
BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
BWRVIP-42-A, "LPCI Coupling Inspection and Flaw Evaluation Guidelines"
BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (see Note)
BWRVIP-100-A, "Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds"
BWRVIP-183, "Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines"

Note: If flaw evaluations are required for BWRVIP-76 examinations, the fracture toughness values of BWRVIP-100-A will be utilized.

Table 1 compares current ASME Examination Category B-N-1 and B-N-2 requirements with the current BWRVIP guideline requirements, as applicable to the Perry Nuclear Power Plant (PNPP). Table 2 provides the inspection history for the PNPP reactor core support structures.

Any deviations from the referenced BWRVIP guidelines for the duration of the proposed alternative will be appropriately documented and communicated to the NRC, per the BWRVIP Deviation Disposition Process. Currently, the PNPP does not have any deviations from the BWRVIP guidelines.

As part of Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Material Issues," BWRs are required to examine reactor internals in accordance with BWRVIP guidelines. These guidelines have been written to address the safety significant vessel internal components and to examine and evaluate the examination results for these components using appropriate methods and re-examination frequencies. The BWRVIP has established a reporting protocol for examination results and deviations. With the exception of BWRVIP-183, which is currently under NRC review, the NRC has agreed with the BWRVIP approach in principle and has issued Safety Evaluations for these guidelines (References 1 – 12). Therefore, use of these guidelines as an alternative to the subject Code requirements provide an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

The Attachment, "Comparison of Code Examination Requirements to BWRVIP Examination Requirements," identifies specific examples that compare the inspection requirements of Table IWB-2500-1, Item Nos. B13.10 and B13.40, to the inspection requirements in the BWRVIP documents. Specific BWRVIP documents are cited as examples. This comparison also includes a discussion of the inspection methods. These comparisons demonstrate that use of these guidelines, as an alternative to the subject Code requirements, provides an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

NRC letter to FirstEnergy Nuclear Operating Company, December 16, 2008, Subject: Perry Nuclear Power Plant, Unit No. 1 – Request for Relief Related to Inservice Inspection Relief Requests Nos. IR-056 and IR-057 (TAC Nos. MD8198 and MD8199).

NRC letter to Entergy Nuclear Operations, September 19, 2005, Subject: Safety Evaluation of Relief Request RI-01, Vermont Yankee Nuclear Power Station (TAC No. MC0690).

8. **References**

1. NRC letter to BWRVIP, June 30, 2008, Subject: Safety Evaluation for Electric Power Research Institute (EPRI) Boiling Water Reactor Vessel and Internals Project (BWRVIP) Report TR-105696-R6 (BWRVIP-03), Revision 6, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines" (TAC No. MC2293).
2. NRC letter to BWRVIP, September 6, 2005, Subject: NRC Approval Letter of BWRVIP-18-A, "BWR Vessel and Internals Project Boiling Water Reactor Core Spray Internals Inspection and Flaw Evaluation Guidelines" (Accession No. ML052490002).
3. NRC letter to BWRVIP, December 19, 1999, Subject: Final Safety Evaluation of BWRVIP Vessel and Internals Project, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)," EPRI Report TR-107284, December 1996 (TAC No. M97802).
4. NRC letter to BWRVIP, September 9, 2005, Subject: NRC Approval Letter of BWRVIP-26-A, "BWR Vessel and Internals Project Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines" (Accession No. ML052490550).
5. NRC letter to BWRVIP, June 10, 2004, Subject: Proprietary Version of NRC Staff Review of BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines."
6. NRC letter to BWRVIP, July 24, 2000, Subject: Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI Report TR-108823 (TAC No. M99638).
7. NRC letter to BWRVIP, February 4, 2001, Subject: Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," (TAC No. M99870).
8. NRC letter to BWRVIP, September 9, 2005, Subject: NRC Approval Letter of BWRVIP-42-A, "BWR Vessel and Internals Project Boiling Water Reactor Low Pressure Coolant Injection and Flaw Evaluation Guidelines" (Accession No. ML052490557).
9. NRC letter to BWRVIP, September 9, 2005, Subject: NRC Approval Letter of BWRVIP-47-A, "BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines" (Accession No. ML052490537).

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10. NRC letter to BWRVIP, July 25, 2005, Subject: NRC Approval Letter of BWRVIP-48-A, "BWR Vessel and Internals Project Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines" (Accession No. ML052130284).
11. NRC letter to BWRVIP, July 27, 2006, Subject: Safety Evaluation of Proprietary EPRI Report , "BWR Vessel and Internals Project, BWR Core Shroud and Inspection and Flaw Evaluation Guidelines (BWRVIP-76)."
12. NRC letter to BWRVIP, November 1, 2007, Subject: NRC Approval Letter with Comment for BWRVIP-100-A, "BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds" (Accession No. ML073050135).
13. BWRVIP-183: BWR Vessel and Internals Project, "Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines," EPRI Technical Report 1013401, December 2007.

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TABLE 1 – Page 1 of 2

Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements for BWR/6 ⁽¹⁾

ASME Item No. Table IWB-2500-1	Core Support Structure Components	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.10	Reactor Vessel Interior	Accessible Areas (Non-specific)	VT-3	Each period	BWRVIP-18-A, 26-A, 38, 41, 42-A, 47-A, 48-A, 76	Overview examinations of components during BWRVIP examinations are performed to satisfy Code VT-3 inspection requirements.		
B13.40	Shroud Support Plate	Accessible Surfaces	VT-3	Each 10-year interval	BWRVIP-38, 3.2.2, Figures 3-4, 3-5	Welds H8 and H9 ⁽²⁾	EVT-1 or UT	Based on as-found conditions, to a maximum 6 years for one side EVT-1, 10 years for UT
	Shroud Support Legs	Accessible Surfaces (beneath core plate; rarely accessible)			BWRVIP-38, 3.2.3	Welds H10, H11 and H12	Per BWRVIP-38 NRC SER (7/24/00), inspect with appropriate method ⁽⁴⁾	When accessible

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TABLE 1 (continued) – Page 2 of 2
Comparison of ASME Examination Category B-N-1 and B-N-2 Requirements With BWRVIP Guidance Requirements for BWR/6 ⁽¹⁾

ASME Item No. Table IWB-2500-1	Core Support Structure Components	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.40	Shroud Horizontal welds	Accessible Surfaces	VT-3	Each 10-year Interval	BWRVIP-76, 2.2 Figure 2-2 ⁽³⁾	Welds H1-H7 as applicable	EVT-1 or UT	Based on as-found conditions, to a maximum 6 years for one side EVT-1, 10 years for UT
	Shroud Vertical welds				BWRVIP-76, 2.3, 3-3, Figures 2-4, 3-2, 3-3	Vertical and Ring Segment Welds	EVT-1 or UT	Maximum 6 years for one-sided EVT-1, 10 years for UT; only required when horizontal welds are found to contain flaws exceeding certain limits or the shroud is a repaired shroud
	Shroud Repairs ⁽³⁾				BWRVIP-76, 3.5, 3.6	Tie-Rod Repair	VT-3	Per repair designer recommendations per BWRVIP-76
	Top Guide and Top Guide Grid				BWRVIP-26-A 3.2 Table 3-2	Top Guide Studs	VT-3	Each 10-year Interval
	Core Support Plate				BWRVIP-25 3.2 Table 3.2	None for BWR/6	N/A	N/A
	Control Rod Guide Tubes (CRGTs)				BWRVIP-47-A 3.2 Table 3.3	CRGT Body Welds and Fuel Support Pins and Lugs	EVT-1 of body welds and VT-3 of pins and lugs	10% of the CRGT Assemblies within 12 years

NOTES:

- 1) This Table provides an overview of the requirements. For more details, refer to ASME Section XI, Table IWB-2500-1, and the appropriate BWRVIP document.
- 2) For Perry, this results in a requirement of 10 percent of the weld length. However, for H9 essentially 100 percent of the weld length was ultrasonically examined.
- 3) Perry's shroud is a Category B un-repaired shroud.
- 4) When inspection tooling and methodologies are available, they will be utilized to establish a baseline inspection of these welds. Until such time, and as committed to in BWRVIP-47-A, Section 3.2.5, visual inspections of the lower plenum area (which includes the shroud support legs) will be performed to the extent practical when access is made available through non-routine refueling outage activities (for example, jet pump disassembly).

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TABLE 2 - Page 1 of 3
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Reactor Core Support Structures Inspection History

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summary of Inspection Results, Repairs, Replacements, Re-inspections
Reactor Vessel Interior (BWRVIP-48-A)	1989 (RF1)	VT-3	In Refueling Outage (RF) 1, VT-3 of 80% of total reactor pressure vessel (RPV) area from the shroud support plate to the flange. The remaining 20% was inaccessible due to the physical lay-out of the Jet Pump (JP) area. There were no relevant indications.
	1992 (RF3)	VT-3 & VT-1	In RF3, VT-3 of the accessible areas was performed along with a VT-1 exam of the vessel wall area near the Feedwater Sparger Spray nozzle ruptures (found during RF3, Nonconformance Report 92-S-045).
	1996 (RF5)	VT-3	In RF5, VT-3 of the accessible areas was performed. There were no relevant indications.
	1999 (RF7)	VT-3	In RF7, VT-3 of the accessible areas was performed. There were no relevant indications.
	2003 (RF9)	VT-3	In RF9, VT-3 of the top head interior was performed. Exam found unusual crud deposits on the upper (i.e., steam region) vessel inside diameter (ID) cladding. Under Condition Report (CR) 03-01995 the hard deposits were evaluated as acceptable for continued operation.
	2007 (RF11)	VT-3	In RF11, VT-3 of 100% of accessible areas above the top guide flange was performed. No indications beyond previously addressed RPV crud.
Core Shroud (BWRVIP-76)	1994 (RF4)	VT-3 & EVT-1	In RF4, VT-3 of entire shroud interior and EVT-1 of the H-3 and H-4 weld inside surfaces at 4 approx. 1-foot long sample locations. No indications.
	1997 (RF6)	VT-3	In RF6, a Code VT-3 exam was performed on all accessible shroud exterior areas. No indications.
	1999 (RF7)	UT	In RF7, UT examination of the H-3, H-4, H-6A and H-7 welds was performed in accordance with the Category B Plant guidelines of BWRVIP-01. No indications.
	2005 (RF10)	UT	In RF10, UT exams of the H-3 and H-4 welds with the Tecnomat ID tool and H-6A and H-7 with the GE OD Tracker. H-4 and H-6A were two sided exams and H-3 and H-7 were one-sided exams. Shallow cracking was found in H-7. It was less than 10% of the inspected length of 67% of the weld and evaluated as acceptable per BWRVIP-76.

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TABLE 2 (continued) - Page 2 of 3
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Reactor Core Support Structures Inspection History

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summary of Inspection Results, Repairs, Replacements, Re-inspections
Shroud Support (BWRVIP-38)	1990 (RF2)	VT-3 & VT-1	In RF2, VT-3 of shroud support plate and VT-1 of the shroud support plate access hole cover. No indications.
	1996 (RF5)	VT-3 & VT-1	In RF5, VT-3 of shroud support plate and VT-1 of the shroud support plate access hole cover. No indications.
	1999 (RF7)	EVT-1	In RF7, baseline EVT-1 exams of the H-8 and H-9 were performed in accordance with BWRVIP-38. No indications.
	2001 (RF8)	VT-1	In RF8, re-seating of JP # 5 provided access to the H-10, H-11 and H-12 welds of the shroud support leg at 90° and approx. 10° of the underside of H-8 and H-9 so they were visually examined with at least VT-1 resolution. No indications.
	2007 (RF11)	EVT-1 & VT-1	In RF11, JP #6 was removed and re-seated due to excess leakage at the transition piece. While disassembled approx. 10° of the underside of H-8 and H-9 were examined with at least VT-1 resolution. Also, the H-10, H-11 and H-12 welds of the shroud support legs at 90° and 120° were examined with EVT-1 resolution. Coverage was approx. 35-50% for the welds of the 90° leg and 25% for the welds of the 120° leg. No indications.
Top Guide (Rim, and so forth) (BWRVIP-26-A)	1989 (RF1)	VT-3	Top Guide periphery, including 90 studs and tack welds, examined in RF1. No indications.
	1994 (RF4)	VT-3	Top Guide grid examined in RF4. No indications.
	1999 (RF7)	VT-1 & VT-3	In RF7, performed VT-3 of the Top Guide assembly in accordance with ASME Category B-N-2 and VT-1 of the studs and tack welds in accordance with BWRVIP-26. No indications.
	2005 (RF10)	VT-3	Code B-N-2 exam of accessible portions of Top Guide grid. Due to ID Core Shroud exams, a significant number of the grid cells were vacated and accessible for inspection. No indications.
Core Plate (Rim, and so forth) (BWRVIP-25; not applicable to BWR/6s)	1989 (RF1)	VT-3	Accessible core plate areas and fuel support castings examined in RF1. No indications.
	1994 (RF4)	VT-3	All of the hold down bolts examined from shroud interior in RF4. No indications.
	1999 (RF7)	VT-3	In RF7, performed VT-3 exam of the core plate areas made accessible by replacement of 5 Control Rod blades in accordance with ASME Category B-N-2. No indications.

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TABLE 2 (continued) - Page 3 of 3
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Reactor Core Support Structures Inspection History

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summary of Inspection Results, Repairs, Replacements, Re-inspections
CRD Guide Tube (BWRVIP-47-A)	1999 (RF7)	VT-1 & EVT-1	In RF7, performed VT-1 of alignment pins and EVT-1 of the welds of 5 Control Rod Guide Tubes in accordance with BWRVIP-47. No indications.
	2001 (RF8)	VT-1 & EVT-1	In RF8, performed VT-1 of alignment pins and EVT-1 of the welds of an additional 4 Control Rod Guide Tubes in accordance with BWRVIP-47 to meet the 5% completion requirements of BWRVIP-47. No indications.
	2005 (RF10)	VT-1 & EVT-1	In RF10, performed VT-1 of alignment pins and EVT-1 of the welds of an additional 5 Control Rod Guide Tubes in accordance with BWRVIP-47. No indications.
	2007 (RF11)	VT-1 & EVT-1	In RF11, performed VT-1 of alignment pins and EVT-1 of the welds of an additional 4 Control Rod Guide Tubes in accordance with BWRVIP-47 to meet the 10% completion (i.e., 18 out of 177) requirements of BWRVIP-47. No indications.
Access Hole Cover (AHC) (BWRVIP-180)	1996 (RF5)	VT-1	VT-1 examination of the access hole cover welds in accordance with SIL-409. No indications.
	2007 (RF11)	EVT-1	EVT-1 examination of the access hole cover welds in accordance with the draft BWRVIP AHC Inspection and Evaluation Guidelines. No indications.

COMPARISON OF CODE EXAMINATION REQUIREMENTS TO BWRVIP EXAMINATION REQUIREMENTS

The following discussion provides a comparison of the examination requirements provided in ASME Section XI, Examination Table IWB-2500-1, Item Nos. B13.10, and B13.40, to the examination requirements in the BWRVIP guidelines. Specific BWRVIP guidelines are cited as examples for comparisons. This comparison also includes a discussion of the examination methods.

Code Requirement - B13.10 - Reactor Vessel Interior Accessible Areas (B-N-1)

The ASME Section XI Code requires a VT-3 examination of reactor vessel accessible areas, which are defined as the spaces above and below the core made accessible during normal refueling outages. The frequency of these examinations is specified as the first refueling outage, and at intervals of approximately three years, during the first inspection interval, and each period during each successive 10-year inspection interval. Typically, these examinations are performed every other refueling outage of the inspection interval. This examination requirement is a non-specific requirement that is a departure from the traditional Section XI examinations of welds and surfaces. As such, this requirement has been interpreted and satisfied differently across the industry. The purpose of the examination is to identify relevant conditions such as: distortion or displacement of parts; loose, missing, or fractured fasteners; foreign material, corrosion, erosion, or accumulation of corrosion products; wear; and structural degradation.

Portions of the various examinations required by the applicable BWRVIP guidelines require access to accessible areas of the reactor vessel during each refueling outage. Examination of core spray piping and spargers (BWRVIP-18-A), top guide (BWRVIP-26-A), jet pump welds and components (BWRVIP-41), interior attachments (BWRVIP-48-A), core shroud welds (BWRVIP-76), shroud support (BWRVIP-38), low pressure coolant injection couplings (BWRVIP-42-A), and lower plenum components (BWRVIP-47-A) provides such access. Locating and examining specific welds and components within the reactor vessel areas above, below (if accessible), and surrounding the core (annulus area) entails access by remote camera systems that essentially perform equivalent VT-3 examination of these areas or spaces as the specific weld or component examinations are performed. This provides an equivalent method of visual examination on a more frequent basis than that required by the ASME Section XI Code. Evidence of wear, structural degradation, loose, missing, or displaced parts, foreign materials, and corrosion product buildup can be, and has been observed during the course of implementing these BWRVIP examination requirements. Therefore, the specified BWRVIP guideline requirements meet or exceed the subject Code requirements for examination method and frequency of the interior of the reactor vessel. Accordingly, these BWRVIP examination requirements provide an acceptable level of quality and safety as compared to the subject Code requirements.

Code Requirement - B13.40 - Core Support Structure (B-N-2)

The ASME Code requires a VT-3 examination of accessible surfaces of the integrally welded core support structure each 10-year interval. In a BWR/6 boiling water reactor, the welded core support structure has primarily been considered the shroud itself and the shroud support structure, including the shroud support plate (annulus floor) the shroud support ring, the shroud support welds, and the shroud support legs (if accessible). Historically, this requirement has been interpreted and satisfied differently across the industry. Category B-N-2 is titled, "Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels." However, since the title for Item No. B13.40 simply states, "Core Support Structure," some plants, including Perry, have also applied the examination requirements to other core support structures such as the control rod guide tubes, core plate and top guide assembly. The proposed alternate examinations replace this ASME requirement with specific BWRVIP guidelines that examine susceptible locations for known relevant degradation mechanisms.

- The Code requires a VT-3 of accessible surfaces each 10-year interval.
- The BWRVIP requires, as a minimum, the same examination method (VT-3) as the Code for integrally welded core support structures, and for specific areas, it requires either an enhanced visual examination technique (EVT-1) or ultrasonic examination (UT).

BWRVIP recommended examinations of core support structures are focused on the known susceptible areas of this structure, including the welds and associated weld heat affected zones. As a minimum, the same or superior visual examination technique is required for examination at the same frequency as the Code examination requirements. In many locations, the BWRVIP guidelines require a volumetric examination of the susceptible welds at a frequency identical to the Code requirement.

The BWRVIP guidelines require an EVT-1 or UT of core support structures. The core shroud, shroud support plate, and top guide grid are used as examples for comparison between the Code and BWRVIP examination requirements as shown below.

Comparison to BWRVIP Requirements - BWR Core Shroud Examination and Flaw Evaluation Guidelines (BWRVIP-76)

- The Code requires a VT-3 examination of accessible surfaces every 10 years.

- BWRVIP-76 requires an EVT-1 examination from the inside and outside surface, where accessible, or UT examination of select circumferential welds that have not been structurally replaced with a shroud repair, at a calculated "end of interval" that will vary depending upon the amount of flaws present, but not to exceed 10 years.

Comparison to BWRVIP Requirements - BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)

- The Code requires a VT-3 examination of accessible surfaces every 10 years.
- The BWRVIP requires examinations of the support plate to shroud weld (H8) and support plate to reactor vessel weld (H9). Examination coverage is required to be (100 percent - Flaw Tolerance) or 10 percent of the weld length, whichever is greater. Examinations are to be performed by EVT-1 or UT from the annulus or UT from the RPV outside surface. Reinspection depends upon the amount of flaws present, but not to exceed six years for EVT-1 or 10 years for UT.

Comparison to BWRVIP Requirements - BWR Shroud Top Guide Grid Inspection and Flaw Evaluation Guidelines (proposed BWRVIP-183)

- The Code requires a VT-3 examination of accessible surfaces every 10 years.
- For BWR/6s, which have top guide grids that are fabricated from two solid plates that are welded together and then the grid is machined out, the BWRVIP requires EVT-1 or UT examinations of the rim areas containing the weld and heat affected zone from the top surface of the top guide and two cells in the same plane/axis as the weld every six years. The regions of the grid beam cells to be inspected are the bottom 2 inches of the interior side surfaces.

In summary, the BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than "all surfaces." The BWRVIP examination methods (EVT-1 or UT) are superior to the Code required VT-3 for flaw detection and characterization. The BWRVIP examination frequency is equivalent to or more frequent than the examination frequency required by the Code. The superior flaw detection and characterization capability, with an equivalent or more frequent examination frequency and the comparable flaw evaluation criteria, results in the BWRVIP criteria providing a level of quality and safety equivalent to or superior to that provided by the Code requirements.

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. American Society of Mechanical Engineers (ASME) Code Components Affected

Reactor Coolant Pressure Boundary Components	Code Class
1B33-F068A/B, Recirculation Pump A/B Discharge Vent Valves	2
1B33-F070A/B, Recirculation Pump A/B Discharge Drain Valves	2
1B33-F065A/B, Recirculation Loop A/B Flow Control Drain Valves	2
1B33-F647A/B, Recirculation Loop A/B Flow Control Vent Valves	2
1B33-F686A/B, Recirculation Loop A/B Flow Control Drain Valves	2
1B33-F027A/B, Recirculation Pump A/B Suction Drain Valves	2
1B33-F503A/B, Instrument Isolation Valves for dPT-NO15A/B	2
1B33-F504A/B, Instrument Isolation Valves for dPT-NO15A/B	2
1B33-F505A, Instrument Isolation Valve for FT-N014C/D	2
1B33-F506A, Instrument Isolation Valve for FT-N014C/D	2
1B33-F505B, Instrument Isolation Valve for FT-N011B and FT-N024C/D	2
1B33-F506B, Instrument Isolation Valve for FT-N011B and FT-N024C/D	2
1B33-F507A, Instrument Isolation Valve for FT-N011A and FT-N014A/B	2
1B33-F508A, Instrument Isolation Valve for FT-N011A and FT-N014A/B	2
1B33-F507B, Instrument Isolation Valve for FT-N024A/B	2
1B33-F508B, Instrument Isolation Valve for FT-N024A/B	2
1B33-F512A/B, Recirculation Pump A/B Differential Pressure Instrument Vent Valves	2
1B33-F513A/B, Recirculation Pump A/B Differential Pressure Instrument Vent Valves	2
1B33-F577, Recirculation Loop B Flow Instrument Vent Valve	2
1B33-F578, Recirculation Loop B Flow Instrument Vent Valve	2
1B33-F579, Recirculation Loop A Flow Instrument Vent Valve	2
1B33-F580, Recirculation Loop A Flow Instrument Vent Valve	2
1B33-F581, Recirculation Loop B Flow Instrument Vent Valve	2
1B33-F582, Recirculation Loop B Flow Instrument Vent Valve	2
1B33-F583, Recirculation Loop A Flow Instrument Vent Valve	2
1B33-F584, Recirculation Loop A Flow Instrument Vent Valve	2
1B33-F059, Recirculation System Sample Isolation Valve	2
1B33-F019, Reactor Water Sample Isolation Valve	2
1B33-F110, Reactor Recirculation Sample Line Drain Valve	2
1G33-F507, Instrument Isolation Valve for FT-N037	2
1G33-F523, Reactor Water Clean-Up (RWCU) Bottom Head Flow Instrument Vent Valve	2
1B21-F596, 1B21-F016 Test Connection Root Valve	2
1B21-F017, Main Steam Drain and Main Steam Isolation Valve Bypass Line Drain Valve	2
1N27-F551A/B/C, Feedwater Header A Branch Test Isolation Valves	2
1N27-F551D/E/F, Feedwater Header B Branch Test Isolation Valves	2

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Reactor Coolant Pressure Boundary Components	Code Class
1N27-F557A/B, Feedwater Header A/B First Test Connection Valves	2
1G33-F508A/B, Instrument Isolation Valves for PT-N076A, PT-N076B	2
1G33-F108, Penetration 131 In-Board Test Connection First Isolation Valve	2
1E31-F540B, RWCU Differential Flow Leak Detection (LD) Low Side Test Connection Valve	2
1E31-F541B, RWCU Differential Flow LD High Side Test Connection Valve	2
1E51-F528A/B/C/D, Instrument Isolation Valves for PT-N084A/B, PT-N085A/B	2
1E31-F542A/B, Reactor Core Isolation Cooling (RCIC)/Residual Heat Removal (RHR) Steam Supply LD Low Standby Test Connection Valves	2
1E31-F543A/B, RCIC/RHR Steam Supply LD High Standby Test Connection Valves	2
1E31-N084B-G, Cross-Tie Low Side PT-N084A/B	2
1E31-N084B-R, Cross-Tie High Side PT-N084A/B	2
1E31-F519, Instrument Isolation Valve for PT-N080A	2
1E31-F545A, RHR A to Low Pressure Core Spray (LPCS) LD High Side Test Connection Valve	2
1E31-F523, Instrument Isolation Valve for PT-N081	2
1E31-F547, High Pressure Core Spray (HPCS) to Standby Liquid Control (SLC) Reference Differential Pressure Test Connection Valve	2
1E31-F520, Instrument Isolation Valve for PT-N080A	2
1E31-F544A, RHR A to LPCS LD Low Side Test Connection Valve	2
1E31-F521, Instrument Isolation Valve for PT-N080B	2
1E31-F522, Instrument Isolation Valve for PT-N080B	2
1E21-F502, LPCS to Reactor Line Test Connection Valve	2
1E22-F501, HPCS to Reactor Line Test Connection Valve	2
1C41-F501, SLC Discharge Line In-Board Drywell Drain Valve	2
1E12-F508A, Low Pressure Coolant Injection (LPCI) From RHR A In-Board First Test Connection Valve	2
1E12-F508B, LPCI From RHR B In-Board First Test Connection Valve	2
1E12-F508C, LPCI From RHR C In-Board First Test Connection Valve	2
1E12-F501, Shutdown Cooling Suction Header In-Board First Connection Valve	2
1B33-F514, Recirculation Jet Pump 15 Flow Instrument Vent Valve	2
1B33-F515, Recirculation Jet Pump 12 Flow Instrument Vent Valve	2
1B33-F516, Recirculation Jet Pump 18 Flow Instrument Vent Valve	2
1B33-F517, Recirculation Jet Pump 19 Flow Instrument Vent Valve	2
1B33-F518, Recirculation Jet Pump 15 Flow Instrument Vent Valve	2
1B33-F519, Recirculation Jet Pump 16 Flow Instrument Vent Valve	2
1B33-F520, Recirculation Jet Pump 11 Flow Instrument Vent Valve	2
1B33-F521, Recirculation Jet Pump 17 Flow Instrument Vent Valve	2
1B33-F522, Recirculation Jet Pump 13 Flow Instrument Vent Valve	2
1B33-F523, Recirculation Jet Pump 20 Flow Instrument Vent Valve	2
1B33-F524, Recirculation Jet Pump 20 Flow Instrument Vent Valve	2
1B33-F525, Recirculation Jet Pump 14 Flow Instrument Vent Valve	2
1B33-F526, Recirculation Jet Pump 15 Flow Instrument Root Valve for FT-N038B, LT-N044D	2

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Reactor Coolant Pressure Boundary Components	Code Class
1B33-F527, Recirculation Jet Pump 12 Flow Instrument Root Valve for FT-N037F	2
1B33-F528, Recirculation Jet Pump 18 Flow Instrument Root Valve for FT-N037M	2
1B33-F529, Recirculation Jet Pump 19 Flow Instrument Root Valve for FT-N037S	2
1B33-F530, Recirculation Jet Pump 15 Flow Instrument Root Valve for FT-N037U, FT-N038B	2
1B33-F531, Recirculation Jet Pump 16 Flow Instrument Root Valve for FT-N037D	2
1B33-F532, Recirculation Jet Pump 11 Flow Instrument Root Valve for FT-N037B	2
1B33-F533, Recirculation Jet Pump 17 Flow Instrument Root Valve for FT-N037H	2
1B33-F534, Recirculation Jet Pump 13 Flow Instrument Root Valve for FT-N037K	2
1B33-F535, Recirculation Jet Pump 20 Flow Instrument Root Valve for FT-N038D	2
1B33-F536, Recirculation Jet Pump 20 Flow Instrument Root Valve for FT-N037W, FT-N038D	2
1B33-F537, Recirculation Jet Pump 14 Flow Instrument Root Valve for FT-N037P	2
1B33-F646, Jet Pump Post Accident Sample Isolation Valve	2
1P87-F001, Reactor Recirculation B Sample Isolation Valve	2
1B33-F538, Recirculation Jet Pump 7 Flow Instrument Vent Valve	2
1B33-F539, Recirculation Jet Pump 9 Flow Instrument Vent Valve	2
1B33-F540, Recirculation Jet Pump 10 Flow Instrument Vent Valve	2
1B33-F541, Recirculation Jet Pump 1 Flow Instrument Vent Valve	2
1B33-F542, Recirculation Jet Pump 2 Flow Instrument Vent Valve	2
1B33-F543, Recirculation Jet Pump 5 Flow Instrument Vent Valve	2
1B33-F544, Recirculation Jet Pump 3 Flow Instrument Vent Valve	2
1B33-F545, Recirculation Jet Pump 10 Flow Instrument Vent Valve	2
1B33-F546, Recirculation Jet Pump 5 Flow Instrument Vent Valve	2
1B33-F547, Recirculation Jet Pump 4 Flow Instrument Vent Valve	2
1B33-F548, Recirculation Jet Pump 6 Flow Instrument Vent Valve	2
1B33-F549, Recirculation Jet Pump 8 Flow Instrument Vent Valve	2
1B33-F550, Recirculation Jet Pump 7 Flow Instrument Root Valve for FT-N037G	2
1B33-F551, Recirculation Jet Pump 9 Flow Instrument Root Valve for FT-N037R	2
1B33-F552, Recirculation Jet Pump 10 Flow Instrument Root Valve for FT-N037V, FT-N038C	2
1B33-F553, Recirculation Jet Pump 1 Flow Instrument Root Valve for FT-N037A	2
1B33-F554, Recirculation Jet Pump 2 Flow Instrument Root Valve for FT-N037E	2

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Reactor Coolant Pressure Boundary Components	Code Class
1B33-F555, Recirculation Jet Pump 5 Flow Instrument Root Valve for FT-N038A, LT-N044C	2
1B33-F556, Recirculation Jet Pump 3 Flow Instrument Root Valve for FT-N037J	2
1B33-F557, Recirculation Jet Pump 10 Flow Instrument Root Valve for FT-N038C	2
1B33-F558, Recirculation Jet Pump 5 Flow Instrument Root Valve for FT-N037T, FT-N038A	2
1B33-F559, Recirculation Jet Pump 4 Flow Instrument Root Valve for FT-N037N	2
1B33-F560, Recirculation Jet Pump 6 Flow Instrument Root Valve for FT-N037C	2
1B33-F561, Recirculation Jet Pump 8 Flow Instrument Root Valve for FT-N037L	2
1B33-F570, Jet Pump Flow Instrument Vent Valve	2
1B33-F571, Jet Pump Flow Instrument Isolation Valve for FT-N037G, FT-N037R, FT-N037V, FT-N037A, FT-N037E, FT-N037J, FT-N037T, FT-N037N, FT-N037C, FT-N037L	2
1B33-F645, Jet Pump Post Accident Sample Isolation Valve	2
1P87-F007, Reactor Recirculation A Sample Isolation Valve	2
1E31-F503, Instrument Isolation Valve for PT-N003A, PT-N086A, PT-N086B	2
1E31-F504, Instrument Isolation Valve for PT-N003A, PT-N086A, PT-N086B	2
1E31-F505, Instrument Isolation Valve for PT-N086C, PT-N086D	2
1E31-F506, Instrument Isolation Valve for PT-N086C, PT-N086D	2
1E31-F507, Instrument Isolation Valve for PT-N003B, PT-N087A, PT-N087B	2
1E31-F508, Instrument Isolation Valve for PT-N003B, PT-N087A, PT-N087B	2
1E31-F509, Instrument Isolation Valve for PT-N087C, PT-N087D	2
1E31-F510, Instrument Isolation Valve for PT-N087C, PT-N087D	2
1E31-F570, Main Steam Line A Flow Instrument Test Connection Valve	2
1E31-F571, Main Steam Line A Flow Instrument Test Connection Valve	2
1E31-F572, Main Steam Line A Flow Instrument Test Connection Valve	2
1E31-F573, Main Steam Line A Flow Instrument Test Connection Valve	2
1E31-F574, Main Steam Line B Flow Instrument Test Connection Valve	2
1E31-F575, Main Steam Line B Flow Instrument Test Connection Valve	2
1E31-F576, Main Steam Line B Flow Instrument Test Connection Valve	2
1E31-F577, Main Steam Line B Flow Instrument Test Connection Valve	2
1E31-F511, Instrument Isolation Valve for PT-N088A, PT-N088B	2
1E31-F512, Instrument Isolation Valve for PT-N088A, PT-N088B	2
1E31-F513, Instrument Isolation Valve for PT-N003C, PT-N088C, PT-N088D	2
1E31-F514, Instrument Isolation Valve for PT-N003C, PT-N088C, PT-N088D	2
1E31-F515, Instrument Isolation Valve for PT-N089A, PT-N089B	2
1E31-F516, Instrument Isolation Valve for PT-N089A, PT-N089B	2
1E31-F517, Instrument Isolation Valve for PT-N003D, PT-N089C, PT-N089D	2
1E31-F518, Instrument Isolation Valve for PT-N003D, PT-N089C, PT-N089D	2
1E31-F578, Main Steam Line C Flow Instrument Test Connection Valve	2
1E31-F579, Main Steam Line C Flow Instrument Test Connection Valve	2
1E31-F580, Main Steam Line C Flow Instrument Test Connection Valve	2

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1E31-F581, Main Steam Line C Flow Instrument Test Connection Valve	2
1E31-F582, Main Steam Line D Flow Instrument Test Connection Valve	2
1E31-F583, Main Steam Line D Flow Instrument Test Connection Valve	2
1E31-F584, Main Steam Line D Flow Instrument Test Connection Valve	2
1E31-F585, Main Steam Line D Flow Instrument Test Connection Valve	2
1B21-F512, Instrument Isolation Valve for LT-N027, LT-N017	2
1B21-F514, Instrument Isolation Valve for LT-N095B, PT-N403B, PI-R004B, PT-N058, PT-N403F, PT-N068B, PT-N008B, PT-N068F, PT-N040, PT-N078B, PT-N062B, PT-N004B, LT-N080B, LT-N490, LT-N091B, LT-N402B, LT-N091F, dPI-R009B, LT-N081B	2
1B21-R011 B-H, Reference Leg Fill Line	2
1B21-R011 B-G, Reference Leg Fill Line	2
1B21-F510, Instrument Isolation Valve for PT-N078D, LT-N080D, LT-N073L, LT-N073R, LT-N081D, LT-N402F, LT-N044D	2
1B21-R011 D-H, Reference Leg Fill Line	2
1B21-R011 D-G, Reference Leg Fill Line	2
1B21-F542, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F511, Instrument Isolation Valve for LT-N080D, dPI-R005	2
1B21-F544, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F546, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F515, Instrument Isolation Valve for LT-N080B, LT-N004, LT-N017, LT-N027, LT-N095B	2
1B21-F551, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F540, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F545, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F509, Instrument Isolation Valve for LT-N073L, LT-N073R, LT-N081D, LT-N402F	2
1B21-F548, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F549, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F513, Instrument Isolation Valve for LT-N081B, LT-N091F, dPI-R009B, LT-N402B, LT-N091B	2
1B21-F583, Instrument Isolation Valve for PT-N081, dPT-N032	2
1B21-F582, Jet Pump Instrument Line Vent Valve	2
1B21-F585, Instrument Isolation Valve For dPT-N011, dPT-N008	2
1B21-F523, Instrument Isolation Valve for Flow Instruments P009, dPI-R005, LT-N490, dPT-N032, FT-N037, FT-N032, dPI-R005	2
1B21-F584, Jet Pump Instrument Line Vent Valve	2
1B21-F553, Instrument Isolation Valve for LT-N095A, PT-N403A, PI-R004A, PT-N403E, PT-N005, PT-N068A, PT-N050, PT-N068E, PT-N006, PT-N008A, PT-N078A, PT-N062A, LT-N004A, LT-N080A, LT-N010, LT-N091A, LT-N402A, dPI-R009A, LT-N091E, LT-N081A	2
1B21-R011 A-H, Reference Leg Fill Line	2
1B21-R011 A-G, Reference Leg Fill Line	2
1B21-F505, Instrument Isolation Valve for LT-N080C, PT-N078C, LT-N004C, LT-N073G, LT-N402E, LT-N073C, LT-N081C, LT-N044C	2
1B21-R011 C-H, Reference Leg Fill Line	2

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1B21-R011 C-G, Reference Leg Fill Line	2
1B21-F536, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F506, Instrument Isolation Valve for LT-N080C, LT-N004C	2
1B21-F539, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F528, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F552, Instrument Isolation Valve for LT-N080A, LT-N004A, LT-N095A	2
1B21-F533, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F535, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F504, Instrument Isolation Valve for LT-N081C, LT-N073C, LT-N402E, LT-N073G	2
1B21-F534, Reactor Pressure Vessel Level Instrument Line Vent Valve	2
1B21-F529, Reactor Pressure Vessel Level Instrument Line Drain Valve	2
1B21-F555, Instrument Isolation Valve for LT-N081A, LT-N091E, dPI-R009A, LT-N402A, LT-N091A, LT-N010	2
1B21-F531, Reactor Pressure Vessel Level Instrument Line Vent Valve	2

2. Applicable Code Edition and Addenda

ASME Section XI, 2001 Edition through the 2003 Addenda

3. Applicable Code Requirement

Table IWC-2500-1, Category C-H, Item No. C7.10 requires all pressure retaining components to be visually examined (VT-2, system leakage test) for evidence of leakage each inspection period. The system pressure test requirements of IWC-5210, which reference IWA-5000 for test conditions required during system leakage tests, also apply. The subject system valves/components are required to operate during normal plant operation; therefore, sub-article IWA-5213(a)(3) requires ASME Class 2 systems to be in operation for at least four hours for insulated components or 10 minutes for noninsulated components prior to commencing system leakage tests.

4. Reason for Request

ASME Class 2 systems are required to be in operation for at least four hours prior to commencing VT-2 examinations. The identified insulated ASME Class 2 valves/components cannot be isolated from the reactor coolant pressure boundary (ASME Class 1). Conducting the ASME Class 2 examinations during the ASME Class 1 system leakage test eliminates the hold time with acceptable quality and safety.

5. Proposed Alternative and Basis for Use

In lieu of IWA-5213(a)(3) and IWC-5210, which requires ASME Class 2 systems to be in operation for at least four hours for insulated components prior to commencing system leakage tests, FENOC proposes to conduct pressure testing in accordance with IWA-5213(a)(1) and IWB-5210, which do not require a hold time.

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For those ASME Class 2 systems/components attached to the reactor coolant pressure boundary (ASME Class 1) that are not provided with either pressure or test isolation, pressure testing would be conducted in accordance with IWA-5213(a)(1) and IWB-5210. That is, components that are required to operate during normal conditions would not be operating for four hours prior to commencing system leakage tests. Instead, the non-isolable (from the ASME Class 1 boundary) ASME Class 2 system valves/components would be examined during the ASME Class 1 system leakage test.

Numerous components attached to the reactor coolant pressure boundary are covered by the provisions of 10 CFR 50.55a(c), Reactor coolant pressure boundary. The piping systems and their associated components connected to the reactor coolant pressure boundary and less than 1 inch in diameter were constructed to the requirements of ASME Section III, Subsection NC, and identified as ASME Class 2 for in-service inspection. The associated components and component parts are identified by valve number and listed above. These piping systems shall be pressurized during the ASME Class 1 reactor coolant pressure boundary system leakage test and a VT-2 visual examination would be performed. The system leakage test frequency and pressure would be that required for an ASME Class 2 system leakage test. Although the system would not have been in operation for four hours prior to commencing the examinations, the time required to bring the reactor coolant system up to test pressure would allow for the detection of leakage.

Within ASME Section XI, the test conditions (that is, pressure, temperature and hold time) between the reactor coolant pressure boundary and other safety systems are different. Although there are differences, the system leakage tests ensure leak tightness. Therefore, the substitution of IWA-5213(a)(1) for IWA-5213(a)(3) and the substitution of IWB-5210 for IWC-5210 satisfies the intent of the Code.

6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the third 10-year in-service inspection interval scheduled to expire May 17, 2019.

7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, November 22, 1999, Subject: Safety Evaluation of the Inservice Inspection Program Second 10-Year Interval Requests for Relief for FirstEnergy Nuclear Operating Company – Perry Nuclear Power Plant, Unit 1 (TAC No. MA3437).