**FENOC**™
*FirstEnergy Nuclear Operating Company*

*Perry Nuclear Power Plant*
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*Frank Payne*
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*440-280-5382*

January 6, 2020
L-19-145

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
<u>10 CFR 50.55a Request in Support of the Fourth 10-Year Inservice Inspection (ISI) Interval</u>

In accordance with the provisions of 10 CFR 50.55a(z)(1), FirstEnergy Nuclear Operating Company (FENOC) hereby requests Nuclear Regulatory Commission (NRC) approval of four proposed alternatives that provide an acceptable level of quality and safety (Enclosures A, B, C, and D).

The Enclosures identify the proposed alternatives, the affected components, the applicable code requirements, the reason for the requests, and the basis for use.

The requests are for use during the fourth 10-year ISI interval, which began on May 18, 2019 and is scheduled to expire May 17, 2029. FENOC is requesting approval of the proposed 10 CFR 50.55a requests by December 31, 2020.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Acting Manager – Nuclear Licensing and Regulatory Affairs, at (330) 315-6808.

Sincerely,

Frank R. Payne

Enclosures:

A. Perry Nuclear Power Plant, 10 CFR 50.55a Request PT-001, Rev. 3
B. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-054, Rev. 2
C. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-056, Rev. 3
D. Perry Nuclear Power Plant, 10 CFR 50.55a Request IR-060, Rev. 0

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager

L-19-145
Enclosure A

Perry Nuclear Power Plant
10 CFR 50.55a Request PT-001, Rev. 3
(7 pages follow)

Proposed Alternative
in Accordance with 10 CFR 50.55a(z)(1)

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

## 1. ASME Code Component(s) Affected

| **Reactor Coolant Pressure Boundary Components** | **Code Class** |
|---|---|
| 1B21-F017, Main Steam Drain and Main Steam Isolation Valve Bypass Line Drain Valve | 2 |
| 1B21-F504, Instrument Isolation Valve for LT-N081C, LT-N073C, LT-N402E, LT-N073G | 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-F506, Instrument Isolation Valve for LT-N080C, LT-N004C | 2 |
| 1B21-F509, Instrument Isolation Valve for LT-N073L, LT-N073R, LT-N081D, LT-N402F | 2 |
| 1B21-F510, Instrument Isolation Valve for PT-N078D, LT-N080D, LT-N073L, LT-N073R, LT-N081D, LT-N402F, LT-N044D | 2 |
| 1B21-F511, Instrument Isolation Valve for LT-N080D, dPI-R005 | 2 |
| 1B21-F512, Instrument Isolation Valve for LT-N027, LT-N017 | 2 |
| 1B21-F513, Instrument Isolation Valve for LT-N081B, LT-N091F, dPI-R009B, LT-N402B, LT-N091B | 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-F515, Instrument Isolation Valve for LT-N080B, LT-N004, LT-N017, LT-N027, LT-N095B | 2 |
| 1B21-F523, Instrument Isolation Valve for Flow Instruments P009, dPI-R005, LT-N490, dPT-N032, FT-N037, FT-N032, dPI-R005 | 2 |
| 1B21-F528, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F529, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F531, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F533, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F534, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F535, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F536, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F539, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F540, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F542, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F544, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F545, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F546, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F548, Reactor Pressure Vessel Level Instrument Line Drain Valve | 2 |
| 1B21-F549, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F551, Reactor Pressure Vessel Level Instrument Line Vent Valve | 2 |
| 1B21-F552, Instrument Isolation Valve for LT-N080A, LT-N004A, LT-N095A | 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 |

| **Reactor Coolant Pressure Boundary Components** | **Code Class** |
|---|---|
| 1B21-F555, Instrument Isolation Valve for LT-N081A, LT-N091E, dPI-R009A, LT-N402A, LT-N091A, LT-N010 | 2 |
| 1B21-F582, Jet Pump Instrument Line Vent Valve | 2 |
| 1B21-F583, Instrument Isolation Valve for PT-N081, dPT-N032 | 2 |
| 1B21-F584, Jet Pump Instrument Line Vent Valve | 2 |
| 1B21-F585, Instrument Isolation Valve For dPT-N011, dPT-N008 | 2 |
| 1B21-F596, 1B21-F016 Test Connection Root Valve | 2 |
| 1B21-R011 A-G, Reference Leg Fill Line | 2 |
| 1B21-R011 A-H, Reference Leg Fill Line | 2 |
| 1B21-R011 B-G, Reference Leg Fill Line | 2 |
| 1B21-R011 B-H, Reference Leg Fill Line | 2 |
| 1B21-R011 C-G, Reference Leg Fill Line | 2 |
| 1B21-R011 C-H, Reference Leg Fill Line | 2 |
| 1B21-R011 D-G, Reference Leg Fill Line | 2 |
| 1B21-R011 D-H, Reference Leg Fill Line | 2 |
| 1B33-F019, Reactor Water Sample Isolation Valve | 2 |
| 1B33-F059, Recirculation System Sample Isolation Valve | 2 |
| 1B33-F065B, Recirculation Loop A/B Flow Control Drain Valves | 2 |
| 1B33-F070B, Recirculation Pump A/B Discharge Drain Valves | 2 |
| 1B33-F110, Reactor Recirculation Sample Line Drain Valve | 2 |
| 1B33-F503A/B, Instrument Isolation Valves for dPT-N015A/B | 2 |
| 1B33-F504A/B, Instrument Isolation Valves for dPT-N015A/B | 2 |
| 1B33-F505A, Instrument Isolation Valve for FT-N014C/D | 2 |
| 1B33-F505B, Instrument Isolation Valve for FT-N011B and FT-N024C/D | 2 |
| 1B33-F506A, Instrument Isolation Valve for FT-N014C/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-F507B, Instrument Isolation Valve for FT-N024A/B | 2 |
| 1B33-F508A, Instrument Isolation Valve for FT-N011A and FT-N014A/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-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 |

| **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-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 |
| 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-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 |

| **Reactor Coolant Pressure Boundary Components** | **Code Class** |
|---|---|
| 1B33-F584, Recirculation Loop A Flow Instrument Vent Valve | 2 |
| 1B33-F645, Jet Pump Post Accident Sample Isolation Valve | 2 |
| 1B33-F646, Jet Pump Post Accident Sample Isolation Valve | 2 |
| 1B33-F686B, Recirculation Loop A/B Flow Control Drain Valves | 2 |
| 1C41-F501, Standby Liquid Control (SLC) Discharge Line In-Board Drywell Drain Valve | 2 |
| 1E12-F501, Shutdown Cooling Suction Header In-Board First Connection Valve | 2 |
| 1E12-F508A, Low Pressure Coolant Injection (LPCI) From Residual Heat Removal (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 |
| 1E21-F502, Low Pressure Core Spray (LPCS) to Reactor Line Test Connection Valve | 2 |
| 1E22-F501, High Pressure Core Spray (HPCS) to Reactor Line Test Connection 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-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-F519, Instrument Isolation Valve for PT-N080A | 2 |
| 1E31-F520, Instrument Isolation Valve for PT-N080A | 2 |
| 1E31-F521, Instrument Isolation Valve for PT-N080B | 2 |
| 1E31-F522, Instrument Isolation Valve for PT-N080B | 2 |
| 1E31-F523, Instrument Isolation Valve for PT-N081 | 2 |
| 1E31-F540B, Reactor Water Clean Up (RWCU) Differential Flow Leak Detection (LD) Low Side Test Connection Valve | 2 |
| 1E31-F541B, RWCU Differential Flow LD High Side Test Connection Valve | 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-F544A, RHR A to LPCS LD Low Side Test Connection Valve | 2 |
| 1E31-F545A, RHR A to LPCS LD High Side Test Connection Valve | 2 |
| 1E31-F547, HPCS to SLC Reference Differential Pressure Test Connection Valve | 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 |

| **Reactor Coolant Pressure Boundary Components** | **Code Class** |
|---|---|
| 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-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 |
| 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 |
| 1E31-N084B-G, Cross-Tie Low Side PT-N084A/B | 2 |
| 1E31-N084B-R, Cross-Tie High Side PT-N084A/B | 2 |
| 1E51-F528A/B/C/D, Instrument Isolation Valves for PT-N084A/B, PT-N085A/B | 2 |
| 1G33-F108, Penetration 131 In-Board Test Connection First Isolation Valve | 2 |
| 1G33-F507, Instrument Isolation Valve for FT-N037 | 2 |
| 1G33-F508A/B, Instrument Isolation Valves for PT-N076A, PT-N076B | 2 |
| 1G33-F523, Reactor Water Clean-Up (RWCU) Bottom Head Flow Instrument Vent 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 |
| 1N27-F557A/B, Feedwater Header A/B First Test Connection Valves | 2 |
| 1N27-F822A/B, Inboard Feedwater Check Valve A/B Inspection First Isolation Valve | 2 |
| 1N27-F824A/B, Outboard Feedwater Check Valve A/B Inspection First Isolation Valve | 2 |
| 1P87-F001, Reactor Recirculation B Sample Isolation Valve | 2 |
| 1P87-F007, Reactor Recirculation A Sample Isolation Valve | 2 |

## 2. Applicable Code Edition

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2013 Edition.

## 3. Applicable Code Requirement(s)

Table IWC-2500-1, Examination 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 and components are required to operate during normal plant operation; therefore, Subarticle IWA-5213(a)(3) requires ASME Class 2 systems to be in operation for

at least four hours for insulated components or ten 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 visual examinations. The identified insulated ASME Class 2 valves and 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, FirstEnergy Nuclear Operating Company (FENOC) proposes to conduct pressure testing in accordance with IWA-5213(a)(1) and IWB-5210, which do not require a hold time.

For those ASME Class 2 systems and 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 and 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 inservice 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 will be performed. The system leakage test frequency and pressure will 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 fourth 10-year inservice inspection interval that expires May 17, 2029.

## 7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, January 31, 2012, Subject: Perry Nuclear Power Plant, Unit No. 1, RE: Safety Evaluation in Support of 10CFR50.55a Requests for the Third 10-Year In-Service Inspection Interval (TAC Nos. ME5373, ME5376, ME5377, ME5379, and ME5380), (ADAMS Accession No ML120180372).

L-19-145
Enclosure B

Perry Nuclear Power Plant
10 CFR 50.55a Request IR-054, Rev. 2
(9 pages follow)

Proposed Alternative
in Accordance With 10 CFR 50.55a(z)(1)

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

## 1. ASME Code Component(s) 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 |

| Reactor Pressure Vessel Welds | Code Class |
|---|---|
| 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 |
| 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

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2013 Edition.

## 3. Applicable Code Requirement(s)

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 would require examination during the fourth inservice 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.

Code Case N-702 proposes that a 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 the conditions placed upon it 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 inservice 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)
For recirculation inlet nozzles: <u>$C_{RPV}$ = 19332 psi</u>
For recirculation outlet nozzles: <u>$C_{RPV}$ = 16171 psi</u>
$C_{NOZZLE}$ = recirculation inlet or outlet nozzles (from BWRVIP-108 model)
For recirculation inlet nozzles: <u>$C_{NOZZLE}$ = 1637 psi</u>
For recirculation outlet nozzles: <u>$C_{NOZZLE}$ = 1977 psi</u>

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

Heatup/Cooldown rate = 100°F/hour
p = reactor pressure vessel (RPV) normal operating pressure, p = 1045 psig (maximum reactor steam dome pressure per Technical Specification 3.4.12)
r = RPV inner radius, r = 120.2"
t = RPV wall thickness, t = 7"
$r_i$ = nozzle inner radius
For inner radius for recirculation outlet (N1) nozzles: $r_i = 10"$
For inner radius for recirculation inlet (N2) nozzles: $r_i = 5.813"$
$r_o$ = nozzle outer radius
For outer radius for recirculation outlet (N1) nozzles: $r_o = 17.594"$
For 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) Maximum RPV Heatup/Cooldown Rate°F/hour

Criterion – the maximum RPV heatup/cooldown rate is limited to < 115°F/hour

In accordance with Technical Specification 3.4.11, reactor coolant system heatup and cooldown rates are maintained at ≤ 100°F in any one hour period.

By letter dated September 17, 2008 (ADAMS 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 (ADAMS Accession No. ML082960729), the NRC staff agreed with PNPP that events in which the 115°F heat-up/cooldown rate was exceeded were transients and as a result the first criterion regarding the maximum RPV heat-up/cooldown rate (which is intended to address normal operating conditions) was satisfied. Since 2008, no other events have occurred where normal heatup/cooldown rates exceeded 115°F/hour.

(2) Recirculation Inlet (N2) Nozzles

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

$(1045 \times 120.2 \div 7) \div 19332 < 1.15$

The PNPP result is 0.93, 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: $(pr/t)/C_{RPV} < 1.15$

$(1045 \times 120.2 \div 7) \div 16171 < 1.15$

The PNPP result is 1.11, 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 fourth 10-year inservice inspection interval that expires May 17, 2029.

## 7. Precedent

Nuclear Regulatory Commission letter to FirstEnergy Nuclear Operating Company, January 31, 2012, Subject: Perry Nuclear Power Plant, Unit No. 1 – RE: Safety Evaluation in Support of 10 CFR 50.55a Requests for the Third 10-Year In-Service Inspection Interval (TAC Nos. ME5373, ME5376, ME5377, ME5379, and ME5380), ADAMS Accession No. ML120180372.

Nuclear Regulatory Commission letter to Exelon Generation Company, LLC, June 28, 2017, Subject: Dresden Nuclear Power Station, Units 2 and 3 – Issuance of Safety Evaluation for Request RE: Inservice Inspection Interval Proposed Alternative (I5R-08) (CAC Nos. MF8090 and MF8091), ADAMS Accession No. ML17073A121.

Nuclear Regulatory Commission letter to Exelon Generation Company, LLC, August 25, 2017, Subject: Quad Cities Nuclear Power Station, Units 1 and 2 – Alternative to the Requirements of the ASME Code Regarding Reactor Pressure Vessel Nozzle Assemblies – Relief Request I5R-07 (CAC Nos. MF8989 and MFMF8990) (RS-16-256), ADAMS Accession No. ML17221A264.

## 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 (ADAMS Accession No. ML073600374).

5. NRC Regulatory Guide 1.147, Revision 18, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.

Table 1

| Component | Outage(s) Inspected - Results* |
|---|---|
| 1B13-N1A-KA | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - NRI |
| 1B13-N1A-IR | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - 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. |
| | 1R16 – One subsurface indication; acceptable. |
| 1B13-N2B-IR | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - 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 |
| | 1R16 - NRI |
| 1B13-N2E-IR | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - 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 |

Table 1 (Continued)

| Component | Outage(s) Inspected - Results* |
|---|---|
| 1B13-N2K-KA | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - NRI |
| 1B13-N2K-IR | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - 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 |
| | 1R17 – NRI |
| 1B13-N3C-IR | 1R5 – NRI |
| | 1R10 – NRI |
| | 1R17 – NRI |
| 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 |
| | 1R17 – NRI |
| 1B13-N5B-IR | 1R5 – NRI |
| | 1R7 – NRI |
| | 1R17 – NRI |

Table 1 (Continued)

| Component | Outage(s) Inspected - Results* |
|---|---|
| 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 |
| | 1R17 – NRI |
| 1B13-N6C-IR | 1R2 – NRI |
| | 1R8 – NRI |
| | 1R17 – NRI |
| 1B13-N7-KA | 1R6 – NRI |
| 1B13-N7-IR | 1R6 – NRI |
| 1B13-N8-KA | 1R1 – NRI |
| | 1R8 – NRI |
| | 1R14 – NRI |
| 1B13-N8-IR | 1R1 – NRI |
| | 1R8 – NRI |
| | 1R14 – NRI |
| 1B13-N9A-KA | 1R2 – NRI |
| 1B13-N9A-IR | 1R2 – NRI |
| 1B13-N9B-KA | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - NRI |
| 1B13-N9B-IR | 1R5 – NRI |
| | 1R11 – NRI |
| | 1R16 - NRI |

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

L-19-145
Enclosure C

Perry Nuclear Power Plant
10 CFR 50.55a Request IR-056, Rev. 3
(13 pages follow)

Proposed Alternative
in Accordance with 10 CFR 50.55a(z)(1)

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

## 1. ASME Code Component(s) Affected

American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Class 1, Examination Category B-N-1 (Interior of Reactor Vessel), and B-N-2 (Welded Core Support Structures and Interior Attachments to Reactor Vessels), Item Numbers:

- B13.10 – Vessel Interior
- B13.20 – Vessel Interior Attachments within Beltline Region
- B13.30 – Interior Attachments beyond Beltline Region
- B13.40 – Core Support Structure

Table 1 provides a detailed list of components associated with each item number.

## 2. Applicable Code Edition

ASME Code, Section XI, 2013 Edition.

## 3. Applicable Code Requirement(s)

ASME Code, Section XI, Paragraph IWB-2500(a) states in part that:

Components shall be examined and tested as specified in Table IWB-2500-1 [“Examination Categories”].

Table IWB-2500-1 specifies, in part, the following visual examinations.

B13.10 Examine accessible areas of the reactor vessel interior each inspection period by the VT-3 visual examination method (B-N-1).

B13.20 Examine accessible interior attachment welds within the beltline region each inspection interval by the VT-1 visual examination method (B-N-2).

B13.30 Examine accessible interior attachment welds beyond the beltline region each inspection interval by the VT-3 visual examination method (B-N-2).

B13.40 Examine accessible surfaces of the core support structure each inspection interval by the VT-3 visual examination method (B-N-2).

## 4. Reason for Request

A wealth of inspection data has been gathered during inspections across the boiling water reactor (BWR) industry. Based on the gathered inspection data, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) has developed inspection and evaluation (I&E) guidelines and has recommended aggressive specific inspection by BWR operators to identify material degradation with BWR components. The BWRVIP guidelines focus on specific and susceptible components, specify appropriate inspection methods capable of identifying known or potential degradation mechanisms, and require re-examination at appropriate intervals. In contrast, the ASME Code inspection requirements were prepared before the BWRVIP initiative and have not evolved with BWR inspection experience.

As an alternative to the ASME Code inspection requirements, use of BWRVIP guidelines will avoid duplicate or unnecessary inspections, while reducing radiological dose.

## 5. Proposed Alternative and Basis for Use

FirstEnergy Nuclear Operating Company (FENOC) proposes to apply the BWRVIP guidelines listed below to affected ASME Code components identified in Table 1, in lieu of the requirements of ASME Code, Section XI, Paragraph IWB-2500(a) and Table IWB-2500-1, including the examination method, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting.

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

BWRVIP-03, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines"

BWRVIP-18, Revision 2-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, Revision 4-A, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"

BWRVIP-42, Revision 1-A, "BWR Vessel and Internals Project, Low Pressure Coolant Injection (LPCI) Coupling Inspection and Flaw Evaluation Guidelines"

BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"

BWRVIP-48, Revision 1, "Vessel ID [Internal Diameter] Attachment Weld Inspection and Flaw Evaluation Guidelines"

BWRVIP-76, Revision 1-A, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (see Note)

BWRVIP-94NP, Revision 3, “BWR Vessel and Internal Project Program Implementation Guide”

BWRVIP-100, Revision 1-A, “Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds”

BWRVIP-138, Revision 1-A, “Updated Jet Pump Beam Inspection and Flaw Evaluation”

BWRVIP-180, “BWR Vessel and Internals Project, Access Hole Cover Inspection and Flaw Evaluation Guidelines”

BWRVIP-183-A, “Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines”

Note: If flaw evaluations are required for BWRVIP-76-R1-A examinations, the fracture toughness values of BWRVIP-100-R1-A will be utilized.

The BWRVIP executive committee periodically revises the BWRVIP guidelines to address industry operating experience, include enhancements to inspection techniques and add or adjust flaw evaluation methodologies. Where the revised version of a BWRVIP inspection guideline continues to also meet the requirements of the version of the BWRVIP inspection guideline that forms the safety basis for the NRC-authorized proposed alternative to the requirements of 10 CFR 50.55a, it may be implemented. Otherwise, the revised guidelines will only be implemented after NRC approval of the revised BWRVIP guidelines or a plant-specific request has been approved.

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 only deviation from BWRVIP guidelines by the PNPP is related to BWRVIP-139-R1-A required examinations on the steam dryer, which is a non-ASME, non-safety related component. Safe and event-free operation of the steam dryer during the extended two-year period is supported by the structural integrity that was observed on the dryer during examinations completed in 2009, 2011, 2013, 2015, and 2017 refueling outages.

Implementation of the proposed alternative actions of this request will be subject to inspection by an Authorized Inspection Agency.

The BWRVIP provides BWR Vessel and Internals Inspection Summaries to the NRC periodically. Reference 20 is the BWRVIP vessel and internals inspection summary transmitted to the NRC that includes PNPP. This summary provides, on a component-by-component basis, the inspection methods utilized, the inspection frequency to date, and the results of the inspections through the spring 2017 outage. This summary also contains the identified corrective actions. Corrective actions and inspections performed prior to the BWRVIP were implemented to the requirements of the ASME Section XI Code, as applicable.

BWRVIP guidelines are written for the safety significant reactor vessel internal components and provide appropriate inspection and evaluation criteria using appropriate methods and re-inspection frequencies. The BWRVIP has established a reporting protocol for inspection results and deviations. The NRC has agreed with the BWRVIP approach (as documented in References 4 through 19 below).

In support of this request, the following information addresses the inspection of furnace-sensitized stainless steel and Alloy 182 welds, specific areas subject to nozzle cracking, and hydrogen water chemistry (HWC) effectiveness.

<u>Furnace-sensitized stainless steel and Alloy 182 welds:</u>
Furnace-sensitized stainless steel vessel attachment welds are inspected as required by the ASME Code and BWRVIP applicable guidelines. The sensitization status of the steam dryer and the feedwater support brackets has not been determined, and as such, they are assumed to be furnace-sensitized.

With regard to Examination Category B-N-1, there are no Alloy 182 welds. With regard to Examination Category B-N-2, the following locations have Alloy 182 welds:

- Shroud support (Weld H9)
- Shroud support legs (Weld H12)
- Access hole cover

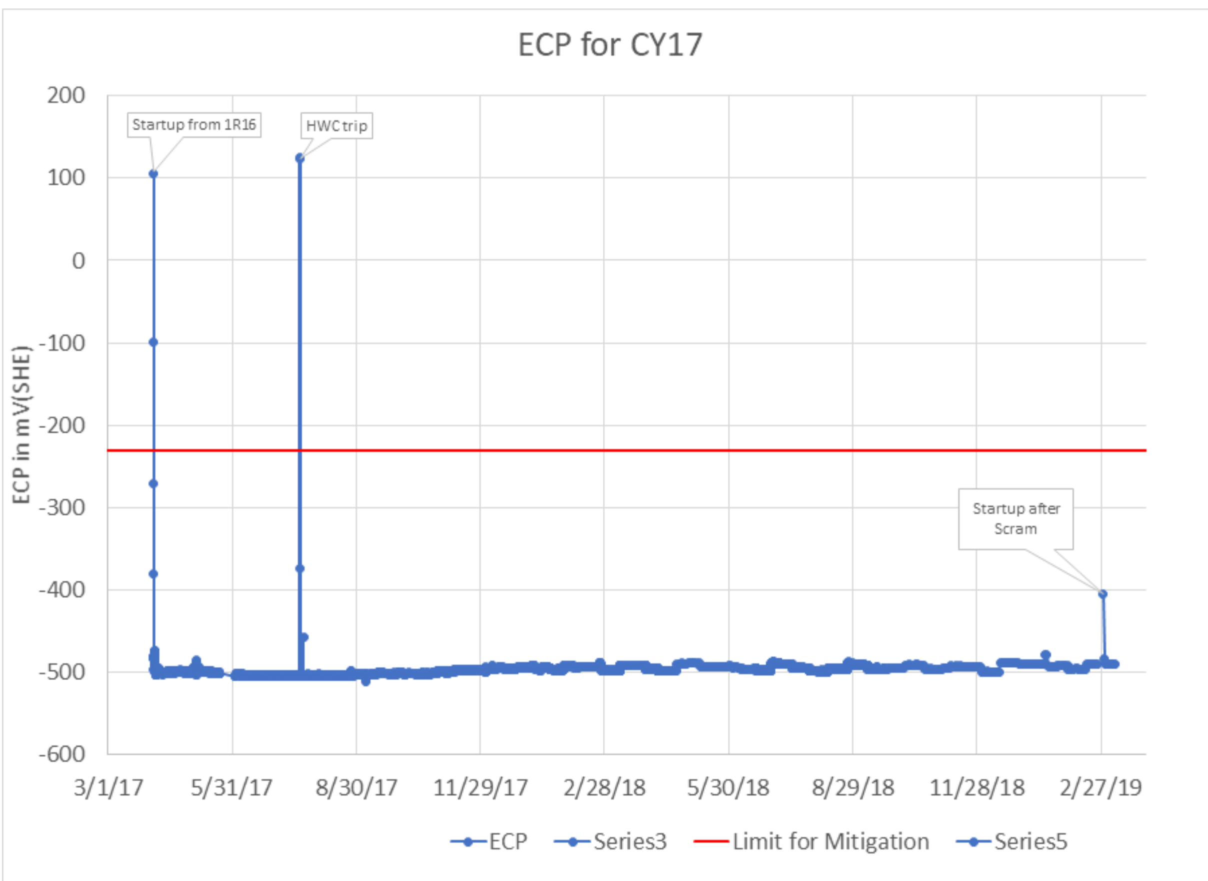
The shroud support and shroud support leg inspections are specified in BWRVIP-38 and the access hole cover inspection is specified in BWRVIP-180. No additional augmented inspections are performed on the Alloy 182 welds outside of that defined in BWRVIP-38 and BWRVIP-180. There has been no cracking identified in the Alloy 182 welds at PNPP.

<u>Feedwater nozzle and control rod drive (CRD) return line nozzle:</u>
NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10 (Technical Report),” is implemented in both the inservice inspection (ISI) and BWRVIP

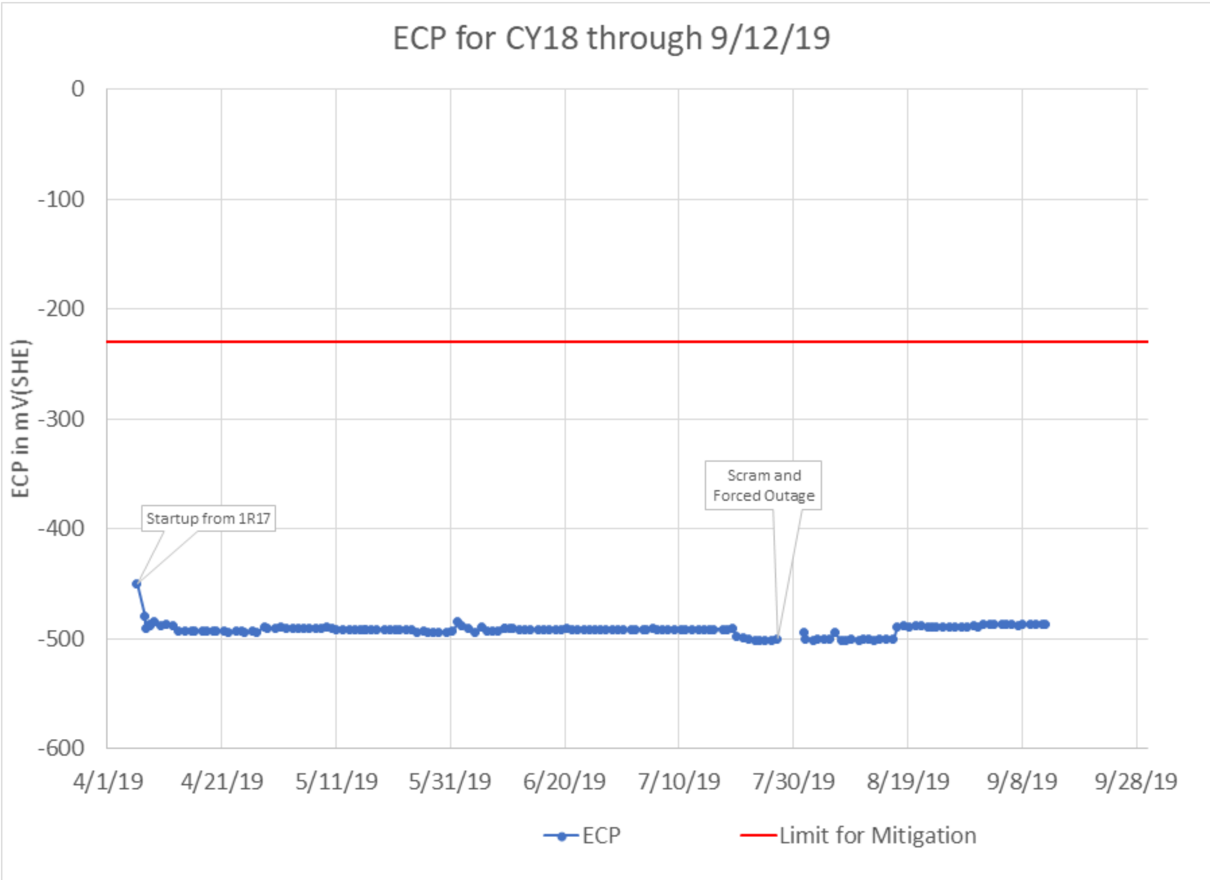
programs as augmented inspections. In regard to the ISI program, the requirements are limited to the six feedwater nozzle inner radii. The CRD return line at the PNPP has been cut and capped and, as such, is not subject to these examinations. The feedwater sparger flow holes and welds in the tees and arms are examined per NUREG-0619. Vessel attachment welds are examined by utilizing the requirements of BWRVIP-48-A.

Effectiveness of hydrogen water chemistry (HWC) in conjunction with noble metal chemical addition (NMCA):
The PNPP utilizes the General Electric (GE) Online NobleChem™ (OLNC) process as the current means of NMCA. This process replaced the previous NobleChem process and is used in conjunction with HWC for the purpose of mitigating intergranular stress corrosion cracking (IGSCC) in the reactor internals. IGSCC is considered mitigated when the local electrochemical corrosion potential (ECP) is reduced below -230 millivolts (mV) standard hydrogen electrode (SHE).

The ECP for Cycle 17 at PNPP was as follows:



The ECP for Cycle 18 (the current cycle of operation) at PNPP is as follows:




The molar ratio value at PNPP is calculated using BWRVIP-202, "BWR Vessel and Internals Application (BWRVIA) for Radiolysis and ECP Analysis Version 3.1." This is an effective tool for estimation of ECP at specific locations and for demonstrating that sufficient hydrogen is being injected to maintain the molar ratio of hydrogen to oxidants greater than 2. Three values are developed at the beginning of an operating cycle: BOC (beginning of cycle), MOC (middle of cycle), and EOC (end of cycle). The values selected are from the upper downcomer location, which is considered the most conservative location by the BWRVIP.

As projected by the BWRVIA model at downcomer, S1 carryunder, the Cycle 17 molar ratio values applicable to the PNPP were:

- BOC: 6.28
- MOC: 5.27
- End of reactivity (before coast down starts): 4.95
- EOC: (coast down): 3.26

As projected by the BWRVIA model at downcomer, S1 carryunder, the Cycle 18 molar ratio values applicable to the PNPP are:

- BOC: 5.92
- MOC: 5.76
- End of reactivity (before coast down starts): 4.70
- EOC: (coast down): 2.65

The needed catalyst loading is > 0.1 μg/cm$^2$. The catalyst loading for the previous two operating cycles at PNPP was as follows:

- Cycle 16 average catalyst loading from in-vessel artifact scraping was 0.97 μg/cm$^2$.
- Cycle 17 average catalyst loading from in-vessel artifact scraping was 0.70 μg/cm$^2$.

HWC availability for the prior and current PNPP operating cycles is as follows:

- Cycle 17 HWC availability was 99.6 percent.
- Cycle 18 HWC availability is currently at 98.3 percent through the end of August 2019. Barring further system trips or scrams, projected availability at end of cycle 18 is 99.6 percent.

Summary

The BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than accessible surfaces. The BWRVIP examination methods (including an enhanced visual examination VT-1 [EVT-1] or ultrasonic test [UT]) are superior to the ASME Code required VT-3 for flaw detection and characterization. In most cases, the BWRVIP examination frequency is equivalent to or more frequent than the examination frequency required by the ASME Code. In cases where the BWRVIP examination frequency is less frequent than required by the ASME Code, the BWRVIP examinations are performed in a more comprehensive manner and focus on the areas most vulnerable. Because the BWRVIP examination methods, frequency, and criteria are developed as a result of industry inspection data, the BWRVIP criteria provides a level of quality and safety that is equivalent or superior to that provided by the ASME Code requirements.

Therefore, use of these BWRVIP guidelines as an alternative to the ASME Code requirements provides an acceptable level of quality and safety.

## 6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the fourth 10-year inservice inspection interval, which expires on May 17, 2029.

## 7. Precedent

Revision 1 of this request, which requested a similar alternative for PNPP's third 10-year inspection interval, was approved by the NRC (Reference 1). Revision 2 of this request for the PNPP third 10-year inspection interval, which sought the approval of the use of additional BWRVIP guidelines, as well as the approval of the use of subsequently revised guidelines, was approved by the NRC (Reference 2).

Requests that permitted the use of BWRVIP-41-R4-A as an alternative to ASME Section XI requirements were approved by the NRC (Reference 3) for Clinton Power Station, Unit 1, James A. Fitzpatrick Nuclear Power Plant, LaSalle County Station, Units 1 and 2, Limerick Generating Station, Units 1 and 2, and Peach Bottom Atomic Power Station, Units 2 and 3.

## 8. References

1) Letter NRC to FirstEnergy Nuclear Operating Company, "Perry Nuclear Power Plant, Unit No. 1, RE: Safety Evaluation in Support of 10 CFR 50.55a Requests for the Third 10-Year In-service Inspection Interval (TAC Nos. ME5373, ME5376, ME5377, ME5379, and ME5380)," dated January 31, 2012 (ADAMS Accession Number ML120180372).

2) Letter NRC to FirstEnergy Nuclear Operating Company, "Perry Nuclear Power Plant, Unit No. 1 – Approval of Alternative to Use BWRVIP Guidelines in Lieu of Certain ASME Code Requirements (CAC No. MG0149; EPID 2017-LLR-0112 (L-17-183)," dated January 29, 2018 (ADAMS Accession No. ML18023A625).

3) Letter NRC to Exelon Generation Company, LLC, "Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; James A. Fitzpatrick Nuclear Power Plant; LaSalle County Station, Units 1 and 2, Limerick Generating Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2 – Revision to Approved Alternatives to Use Boiling Water Reactor Vessel and Internals Project Guidelines (EPID L-2019-LLR-0012)," dated April 30, 2019 (ADAMS Accession No. ML19098A034).

4) Letter NRC to BWRVIP, "Clarification to NRC Review of BWRVIP-03," dated September 21, 2001 (ADAMS Accession No. ML012670353).

5) Letter NRC to BWRVIP, "Final Safety Evaluation for Electric Power Research Institute Topical Report 'BWRVIP-18, Revision 2: Boiling Water Reactor Vessel and Internals Project, Boiling Water Reactor Vessel Core Spray Internals Inspection and Flaw Evaluation Guidelines (TAC No. MF8809),'" dated February 22, 2016 (ADAMS Accession No. ML16011A190).

6) Letter NRC to BWRVIP, "Final Safety Evaluation of BWRVIP Vessel and Internals Project, BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25)," EPRI Report TR-107284, December 1996 (TAC No. M97802), dated December 19, 1999.

7) Letter NRC to BWRVIP, "NRC Approval Letter of BWRVIP-26-A, 'BWR Vessel and Internals Project Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines,'" dated September 9, 2005 (ADAMS Accession No. ML052490550).

8) Letter NRC to BWRVIP, "Proprietary version of NRC Staff Review of BWRVIP-27-A, 'BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines,'" dated June 10, 2004.

9) Letter NRC to BWRVIP, "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)" dated July 24, 2000.

10) Letter NRC to BWRVIP, "Final Proprietary Safety Evaluation for Electric Power Research Institute Topical Report BWRVIP-41, Revision 4, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines" (CAC No. MF4887; EPIC L-2014-TOP-0008)," dated June 26, 2018 (ADAMS Accession No. ML18130A050).

11) Letter NRC to BWRVIP, "US. Nuclear Regulatory Commission Approval and Proprietary Determination Letter for Electric Power Research Institute "BWRVIP-42, Revision 1-A: BWR Vessel and Internals Project, Low Pressure Coolant Injection (LPCI) Coupling Inspection and Flaw Evaluation Guidelines" (EPIC: L-2018-TOP-0010)," dated April 16, 2008 (ADAMS Accession No. ML17004A256).

12) Letter NRC to BWRVIP, "NRC Approval Letter of BWRVIP-47-A, 'BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines,'" dated September 9, 2005.

13) Letter NRC to BWRVIP, "NRC Approval Letter of BWRVIP-48, Revision 1 "Boiling Water Reactor Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, EPRI Technical report 3002013094," dated June 2019.

14) Letter NRC to BWRVIP, "Final Safety Evaluations of the Boiling Water Reactor Vessel and Internals Project 76, Rev. 1-A Topical Report, Boiling Water Reactor Core Shroud Inspection and Flaw Evaluation Guidelines (TAC No. ME8317)," dated November 12, 2014.

15) BWRVIP-94NP, Revision 3: BWR Vessel and Internals Project, Program Implementation Guide, dated September 2018.

16) Letter NRC to BWRVIP, "Final Proprietary Safety Evaluation for Electric Power Research Institute Topical Report BWRVIP-100, Revision 1, 'BWRVIP Vessel and Internals Project: Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds' (TAC No. ME8329)," dated April 12, 2016.

17) Letter NRC to BWRVIP, "Electric Power Research Institute Final Safety Evaluation for Technical Report 1016574 'BWRVIP-138, Revision 1-A: BWR [Boiling Water Reactor] Vessel and Internals Project: Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines' (TAC No. ME2191)," dated May 14, 2012.

18) "BWRVIP-180: BWR Vessel and Internals Project, Access Hole Cover Inspection and Flaw Evaluation Guidelines," dated November 2007.

19) Letter NRC to BWRVIP, "Final Safety Evaluation for Electric Power Research Institute Topical Report BWRVIP-183-A, 'BWR Vessel and Internals Project, Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines' (TAC No. ME2178)," dated December 31, 2015.

20) Letter 2018-015 from BWRVIP to NRC, "Project No. 704 – BWR Vessel and Internals Inspection Summaries for Spring 2017 Outages," dated February 7, 2018 (ADAMS Accession No. ML18040A464).

**TABLE 1 – Page 1 of 3**

**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 | Components | ASME Exam Scope | ASME Exam | ASME Frequency | Applicable BWRVIP Document | BWRVIP Exam Scope | BWRVIP Exam | BWRVIP Frequency |
|---|---|---|---|---|---|---|---|---|
| B13.10 Reactor Vessel Interior | Reactor Vessel Interior | Accessible Areas (Non-specific) | VT-3 | Each period | BWRVIP-18-R2-A, 26-A, 38, 41-R4-A, 42-R1-A, 47-A, 48-A, 76-R1-A | Overview examinations of components during BWRVIP examinations are performed to satisfy ASME Code VT-3 inspection requirements. | | |
| B13.20 Interior Attachments Within Beltline Region | Jet Pump Riser Brace | Accessible welds | VT-1 | Each 10-year Interval | BWRVIP-48-A Table 3-2 | Jet pump riser brace bracket welds | EVT-1 | 25% every 6 years |
| | RPV Surveillance Sample Holder | | | | | Surveillance sample holder bracket welds | VT-1 | Each 10-Year interval |
| B13.30 Interior Attachments Beyond Beltline | Core Spray piping bracket | Accessible welds | VT-3 | Each 10-year Interval | BWRVIP-48-A Table 3-2 | Core Spray primary and supplemental bracket | EVT-1 | 100% every four refueling cycles |
| | Feedwater sparger bracket | | | | | Feedwater sparger bracket attachment welds | EVT-1 | Each 10-Year interval |
| | Guide Rod support bracket | | | | | Guide rod support bracket welds | VT-3 | Each 10-Year interval |
| | Steam Dryer hold down bracket | | | | | Steam Dryer hold down bracket welds | VT-3 | Each 10-Year interval |
| | Steam Dryer support bracket | | | | | Steam Dryer support bracket welds | EVT-1 | Each 10-Year interval |

**TABLE 1 (continued) – Page 2 of 3**

**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 | Components | ASME Exam Scope | ASME Exam | ASME Frequency | Applicable BWRVIP Document | BWRVIP Exam Scope | BWRVIP Exam | BWRVIP Frequency |
|---|---|---|---|---|---|---|---|---|
| B13.40 Core Support Structure | 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 of 6 years for one sided 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 |
| | Shroud Horizontal welds | Accessible Surfaces | VT-3 | Each 10-year Interval | BWRVIP-76-R1-A, 2.2 Figure 2-2[(3)] | Welds H1-H7 as applicable | EVT-1 or UT | Based on as-found conditions, to a maximum of 6 years for one sided EVT-1, 10 years for UT |
| | Shroud Vertical welds | | | | BWRVIP-76-R1-A, 2.3, 3.3, Figures 2-4, 3-2, 3-3 | Vertical and Ring Segment Welds | EVT-1 or UT | Maximum of 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 |

**TABLE 1 (continued) – Page 3 of 3**

**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 | Components | ASME Exam Scope | ASME Exam | ASME Frequency | Applicable BWRVIP Document | BWRVIP Exam Scope | BWRVIP Exam | BWRVIP Frequency |
|---|---|---|---|---|---|---|---|---|
| B13.40 Core Support Structure | Shroud Repairs[(3)] | Accessible Surfaces | VT-3 | Each 10-year Interval | BWRVIP-76-R1-A, 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 only an overview of the requirements. For more details, refer to the ASME Code, Section XI, Table IWB-2500-1, and the appropriate BWRVIP document.
2) For PNPP this results in a requirement of 10% of the weld length. However, for H9 essentially 100 percent of the weld length was ultrasonically examined.
3) PNPP'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 (e.g., jet pump disassembly).

L-19-145
Enclosure D

Perry Nuclear Power Plant
10 CFR 50.55a Request IR-060, Rev. 0
(12 pages follow)

Proposed Alternative
in Accordance With 10 CFR 50.55a(z)(1)

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

## 1. ASME Code Component(s) Affected

| | |
|---|---|
| Component Number: | 72 reactor pressure vessel (RPV) threads in flange 1B13-A1-T to 1B13-A9-T; 1B13-B1-T to 1B13-B9-T; 1B13-C1-T to 1B13-C9-T; 1B13-D1-T to 1B13-D9-T; 1B13-E1-T to 1B13-E9-T; 1B13-F1-T to 1B13-F9-T; 1B13-G1-T to 1B13-G9-T; and 1B13-H1-T to 1B13-H9-T |
| Code Class: | 1 |

## 2. Applicable Code Edition

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2013 Edition.

## 3. Applicable Code Requirement(s)

Table IWB-2500-1, "Examination Category B-G-1, Pressure Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter," Item No. B6.40, requires volumetric examination of reactor pressure vessel (RPV) threads in flange (that is, threads in 100 percent of the flange stud holes), each successive inspection interval. The examination area is the one-inch area around each RPV stud hole, as shown on Figure IWB-2500-12, "Closure Stud and Threads in Flange Stud Hole."

## 4. Reason for Request

In accordance with 10 CFR 50.55a(z)(1), FENOC is requesting to use ASME Code Case N-864 as an alternative to eliminate the examination of threads in RPV flanges required by Examination Category B-G-1, Item No. B6.40 of the ASME Code, Section XI, for the remainder of the duration of the fourth 10-year ISI interval.

## 5. Proposed Alternative and Basis for Use

In lieu of the ISI requirements for a volumetric examination of RPV threads in flanges, required by Table IWB-2500-1, "Examination Category B-G-1, Pressure Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter," Item No. B6.40, FENOC proposes to use ASME Code Case N-864.

To protect against non-service related degradation, FENOC uses detailed procedures for the care and visual inspection of the RPV studs (also, herein referred to as bolts)

and the threads in flange each time the RPV closure head is removed at the Perry Nuclear Power Plant (PNPP). Care is taken to inspect the RPV threads for damage and to protect threads from damage when the studs are removed. Prior to reinstallation, the studs and stud holes are cleaned and lubricated. The studs are then re-inserted and tensioned into the RPV flange. This activity is performed each time the closure head is removed, and the procedure documents each step. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

Electric Power Research Institute (EPRI) Report No. 3002007626, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements" (Reference 1, hereafter referred to as the EPRI report), provides an acceptable technical basis for elimination of the examination requirement. The EPRI report includes:

- A review of operating experience related to RPV flange bolting including a survey of inspection results from 168 units,
- An evaluation of the susceptibility of the threads in the RPV flange to potential degradation mechanisms,
- A stress analysis to determine the stresses at the critical locations of the component,
- A flaw tolerance evaluation to determine how long it would take a postulated flaw to challenge the integrity of the RPV.

The technical basis for the proposed alternative is discussed in more detail below.

<u>Operating Experience Review</u>

A survey of inspection results, which includes results from PNPP, confirmed that the RPV threads in flange examination are adversely impacting outage activities (worker exposure, personnel safety, and critical path time) while not identifying any service induced degradations. Specifically, for the U.S. fleet, a total of 94 units have responded to date and none of these units have identified any type of degradation. As can be seen in Table 1 below, the data is encompassing. The 94 units represent data from 33 boiling water reactors (BWRs) and 61 pressurized water reactors (PWRs). For the BWR units, a total 3,793 examinations were conducted and for the PWR units a total of 6,869 examinations were conducted, with no service-induced degradation identified. The response data includes information from the plant designs in operation in the U.S. and includes BWR-2, -3, -4, -5, and -6 designs. The PWR plants include the 2-loop, 3-loop, and 4-loop designs and each of the PWR nuclear steam supply system (NSSS) designs (that is, Babcock & Wilcox, Combustion Engineering, and Westinghouse).

**Table 1: Summary of Survey Results – U.S. Fleet**

| Plant Type | Number of Units | Number of Examinations | Number of Reportable Indications |
|---|---|---|---|
| BWR | 33 | 3,793 | 0 |
| PWR | 61 | 6,869 | 0 |
| Total | 94 | 10,662 | 0 |

The EPRI report indicates that many studies were conducted in response to the anticipated transient without scram (ATWS) rule issued by the Nuclear Regulatory Commission (NRC). The rule was issued to require design changes to reduce expected ATWS frequency and consequences. The studies were conducted to understand the ATWS phenomena and key contributors to successful response to an ATWS event. The reactor coolant system (RCS) and its individual components were reviewed to determine weak links. The key take-away for these studies is that the RPV flange ligament was not identified as a weak link. Thus, there is substantial structural margin associated with the RPV flange.

In summary, the EPRI report identifies that the RPV threads in flange are performing with very high reliability based on operating and examination experience. This is due to the robust design and a relatively benign operating environment (for example, the number and magnitude of transients is small, generally not in contact with primary water at plant operating temperatures and pressures). The robust design is reflected by the fact that plant operation has been allowed at several plants even with a reactor closure head bolt assumed to be out of service.

Potential Degradation Mechanisms

An evaluation of potential degradation mechanisms that could impact RPV threads in flange reliability was performed as part of the EPRI report. Potential types of degradation evaluated included pitting, intergranular attack, corrosion fatigue, stress corrosion cracking, crevice corrosion, velocity phenomena, dealloying corrosion, general corrosion, stress relaxation, creep, mechanical wear, mechanical fatigue, and thermal fatigue. Other than the potential for mechanical and thermal fatigue, there are no active degradation mechanisms identified for the threads in flange component.

The EPRI report notes a general conclusion from Reference 2 (which includes work supported by the NRC) that when a component item has no active degradation mechanism present, and a preservice inspection has confirmed that the inspection volume is in good condition (that is, contains no flaws or indications), then subsequent inservice inspections do not provide additional value going forward. As discussed in the Operating Experience review above, the RPV flange ligaments have received the required preservice examinations and over 10,000 inservice inspections, with no relevant findings.

Stress Analysis

To address the potential for mechanical and thermal fatigue, the EPRI report documents a stress analysis and flaw tolerance evaluation of the RPV threads in flange component. The evaluation consists of two parts. In the first part, a stress analysis is performed considering applicable loads on the threads in flange component. In the second part, the stresses at the critical locations of the component are used in a fracture mechanics evaluation to determine the allowable flaw size for the component as well as how much time it will take for a postulated initial flaw to grow to the allowable flaw size using guidelines in ASME Code Section XI, IWB-3500.

The PWR design was selected as a representative geometry for the finite element model because of its higher design pressure and temperature. This representative geometry used the largest PWR RPV diameter along with the largest bolts and the highest number of bolts. The larger and more numerous bolt configuration results in less flange material between bolt holes, whereas the larger RPV diameter results in higher pressure and thermal stresses.

Sixteen nuclear plant units (10 PWRs and 6 BWRs) were considered in the analysis. The evaluation was performed using a geometric configuration that bounds the 16 units considered in this effort. The details of the RPV parameters for PNPP as compared to the bounding values used in the evaluation are shown in Table 2. The preload stress for PNPP is bounded by the EPRI report. Specifically, the EPRI report preload stress is 42,338 pounds per square inch (psi) whereas the preload stress is 28,143 psi at PNPP. The PNPP stress is bounded by the EPRI report, which demonstrates that the report remains applicable to this relief request.

For comparison purposes, the global force per flange stud can be estimated by the pressure force on the flange (p*π*r2, where p is the design pressure and r is the vessel inside radius at the stud hole elevation) divided by the number of stud holes. From the parameters in Table 2, this results in a value of 1,088 kips per stud for the configuration used in the analysis and 775.6 kips per stud for the PNPP configurations, indicating that the configuration used in the analysis bounds that at PNPP.

The specifications for the threads and thread geometry for PNPP as compared to that used in the analysis in EPRI report is shown in Table 3. As this table shows, the flange hole diameter used in the analysis is greater than the PNPP flange hole diameter. The larger hole diameter results in a smaller remaining ligament between holes and is therefore conservative. As can be seen from Table 3, the pitch of the threads used in the analysis is identical to the pitch of the threads for PNPP. For PNPP, the depth of the threads is slightly larger than that used in the analysis. Deeper threads result in lower stress; therefore, the analysis is conservative. Hence the thread geometry used in the analysis is representative of the thread geometry for PNPP. Dimensions of the analyzed geometry are shown in Figure IR-060-1.

**Table 2: Parameters Compared to Bounding Values Used in Analysis**

| Plant | No. of Studs Currently Installed | Stud Nominal Diameter (inches) | RPV Inside Diameter at Stud Hole (inches) | Flange Thickness at Stud Hole (inches) | Design Pressure (psig) | Preload Stress (psi) |
|---|---|---|---|---|---|---|
| PNPP | 72 | 6.25 | 238.5 | 14.3125 | 1250 | 28,143 |
| Range for 16 Units Considered | 54 - 76 | 6 - 7 | 155 - 250 | 12.9 – 16 | 2500 | 42,338 |
| **Bounding Values Used in Analysis** | **54** | **6.0** | **173** | **16** | **2500** | **NA** |

**Table 3: RPV Flange Thread Geometry**

| Plant | Thread Specification | Nominal Bolt Hole Diameter in Flange (inches) | Pitch | Thread Depth (inches) |
|---|---|---|---|---|
| PNPP | 6.25"-8UN-2B | 6.25 | 8 | 0.06765 |
| **Analysis Geometry per EPRI Report** | **7"-8N-2B** | **7.00** | **8** | **0.06500** |

The analytical model is shown in Figures IR-060-2 and IR-060-3. The loads considered in the analysis consisted of:

- A design pressure of 2500 psi gauge at an operating temperature of 600ºF was applied to surfaces exposed to internal pressure.
- Bolt preload – Stress of 42,338 psi.
- Thermal stresses - The only significant transient affecting the bolting flange is heat-up/cooldown. This transient typically consists of a steady 100ºF per hour ramp up to the operating temperature, with a corresponding pressure ramp up to the operating pressure.

The ANSYS finite element analysis program was used to determine the stresses in the thread in flange component for the three loads described above.

Flaw Tolerance Evaluation

A flaw tolerance evaluation was performed using the results of the stress analysis to determine how long it would take an initial postulated flaw to reach the ASME Section XI allowable flaw size. A linear elastic fracture mechanics evaluation consistent with ASME Section XI, IWB-3600 was performed.

Stress intensity factors (K's) at four flaw depths of a 360-degree inside-surface-connected, partial-through-wall circumferential flaws are calculated using finite element analysis techniques with the model described above. The maximum stress intensity factor (K) values around the bolt hole circumference for each flaw depth (a) are extracted and used to perform the crack growth calculations. The circumferential flaw is modeled to start between the tenth and eleventh flange threads from the top end of the flange because that is where the largest tensile axial stress occurs. The modeled flaw depth–to–wall thickness ratios (a/t) are 0.02, 0.29, 0.55, and 0.77, as measured in any direction from the stud hole. This creates an ellipsoidal flaw shape around the circumference of the flange, as shown in Figure IR-060-4 for the flaw model with a/t = 0.77 a/t crack model. The crack tip mesh for the other flaw depths follows the same pattern. When preload is not being applied, the stud, stud threads, and flange threads are not modeled. The model is otherwise unchanged between load cases.

The maximum K results are summarized in Table 4 for the four crack depths. From Table 4, the maximum K occurs at operating conditions (preload + heatup + pressure). Because the crack tip varies in depth around the circumference, the maximum K from all locations at each crack size is conservatively used for the K vs. a profile.

**Table 4: Maximum K vs. a/t**

| Load | K at Crack Depth (ksi√in) | | | |
|---|---|---|---|---|
| | 0.02 a/t | 0.29 a/t | 0.55 a/t | 0.77 a/t |
| Preload | 11.2 | 17.4 | 15.5 | 13.9 |
| Preload + Heatup + Pressure | 13.0 | 19.8 | 16.1 | 16.3 |

The allowable stress intensity factor was determined as shown below, based on the acceptance criteria in ASME Section XI, IWB-3612.
$K_I < K_{Ic}/\sqrt{10} = 69.6$ ksi√in

Where,
$K_I$ = Allowable stress intensity factor (ksi√in)
$K_{Ic}$ = Lower bound fracture toughness at operating temperature (220 ksi√in)

As can be seen from Table 4, the allowable stress intensity factor is not exceeded for all crack depths up to the deepest analyzed flaw of a/t = 0.77. Hence the allowable flaw depth of the 360-degree circumferential flaw is at least 77 percent of the thickness of the flange.

For the crack growth evaluation, an initial postulated flaw size of 0.2 in. (5.08 mm) is chosen consistent with the ASME Code Section XI, IWB-3500 flaw acceptance standards. The deepest flaw analyzed is a/t = 0.77 because of the inherent limits of the model. Two load cases are considered for fatigue crack growth: heat-up and cooldown, and bolt preload.

The heat-up and cooldown load case includes the stresses due to thermal and internal pressure loads and is conservatively assumed to occur 50 times per year. The bolt preload is assumed to be present and constant during the load cycling of the heat-up and cooldown load case. The bolt preload load case is conservatively assumed to occur five times per year, and these cycles do not include thermal or internal pressure. The resulting crack growth was determined to be negligible due to the small delta K and the relatively low number of cycles associated with the transients evaluated. Because the crack growth is insignificant, the allowable flaw size will not be reached, and the integrity of the component is not challenged for at least 80 years (original 40-year design life plus additional 40 years of plant life extension).

An evaluation was also performed to determine the acceptability at preload condition. Table 5 below provides the RPV flange $RT_{NDT}$ values and the bolt-up temperatures for PNPP. These were determined using the $RT_{NDT}$ value from plant records. As can be seen from this table, the minimum ($T-RT_{NDT}$) is 80°F, corresponding to PNPP. From the equations in paragraph A-4200 of ASME Section XI, Appendix A, the corresponding values of $K_{Ic}$ are 135.9 ksi√in. Using a structural factor of √10, the allowable $K_{Ic}$ value is 43.0 ksi√in. This value is more than the maximum stress intensity factor ($K_I$) for the preload condition of 17.4 ksi√in shown in Table 4, thus the report evaluation is bounding for the PNPP.

**Table 5: RPV Flange $RT_{NDT}$ and Bolt-Up Temperature**

| Plant Name | Flange $RT_{NDT}$ (°F) | Preload Temp (°F) | Minimum $T-RT_{NDT}$ (°F) |
|---|---|---|---|
| PNPP | -10 | >70 | 80 |

The stress analysis and flaw tolerance evaluation presented above shows that the thread in flange component is very flaw tolerant and can operate for 80 years without violating ASME Section XI safety margins. This clearly demonstrates that the thread in flange examinations can be eliminated without affecting the safety of the RPV.

ASME Code Case N-864

Request IR-060, Revision 0, proposes that the volumetric examination of RPV threads in flange (required by Table IWB-2500-1, Item No. B6.40) is not required, and this proposed alternative is consistent with the opinion stated in ASME Code Case N-864 (Reference 3). The affected ASME Code components identified in Request IR-060 are components within the scope of ASME Code Case N-864.

ASME Code Case N-864 was approved by the American Society of Mechanical Engineers Board on Nuclear Codes and Standards on July 28, 2017; however, it has not been incorporated into NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and is not available for application at nuclear power plants without specific NRC approval.

Conclusion

Since there is reasonable assurance that the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, authorization to use the proposed alternative is requested in accordance with 10 CFR 50.55a(z)(1) on the basis that use of the alternative provides an acceptable level of quality and safety.

## 6. Duration of Proposed Alternative

This proposed alternative shall be utilized during the fourth 10-year inservice inspection interval, which is currently scheduled to expire May 17, 2029.

## 7. Precedent

The NRC authorized the use of a similar request to eliminate the examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, threads in the reactor pressure vessel flange for Exelon Generation Company, LLC (EGC) by letter dated June 26, 2017 (Accession No. ML17170A013). This request was part of an EGC fleet submittal, and the alternative was authorized for various stations. No changes were made for Category B-G-1, Item Number B6.40, components between the ASME Code edition and addenda used for these requests and the 2013 Edition used by PNPP.

## 8. References

1. Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements. EPRI, Palo Alto, CA: 2016. 3002007626 (ADAMS Accession No. ML16221A068).

2. American Society of Mechanical Engineers, Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, Light Water Reactor (LWR) Nuclear Power Plant Components. CRTD-Vols. 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.

3. ASME Boiler and Pressure Vessel Code, Code Case N-864, "Reactor Vessel Threads in Flange Examinations, Section XI, Division 1," approved July 28, 2017.

**Figure IR-060-1**
**Modeled Dimensions**

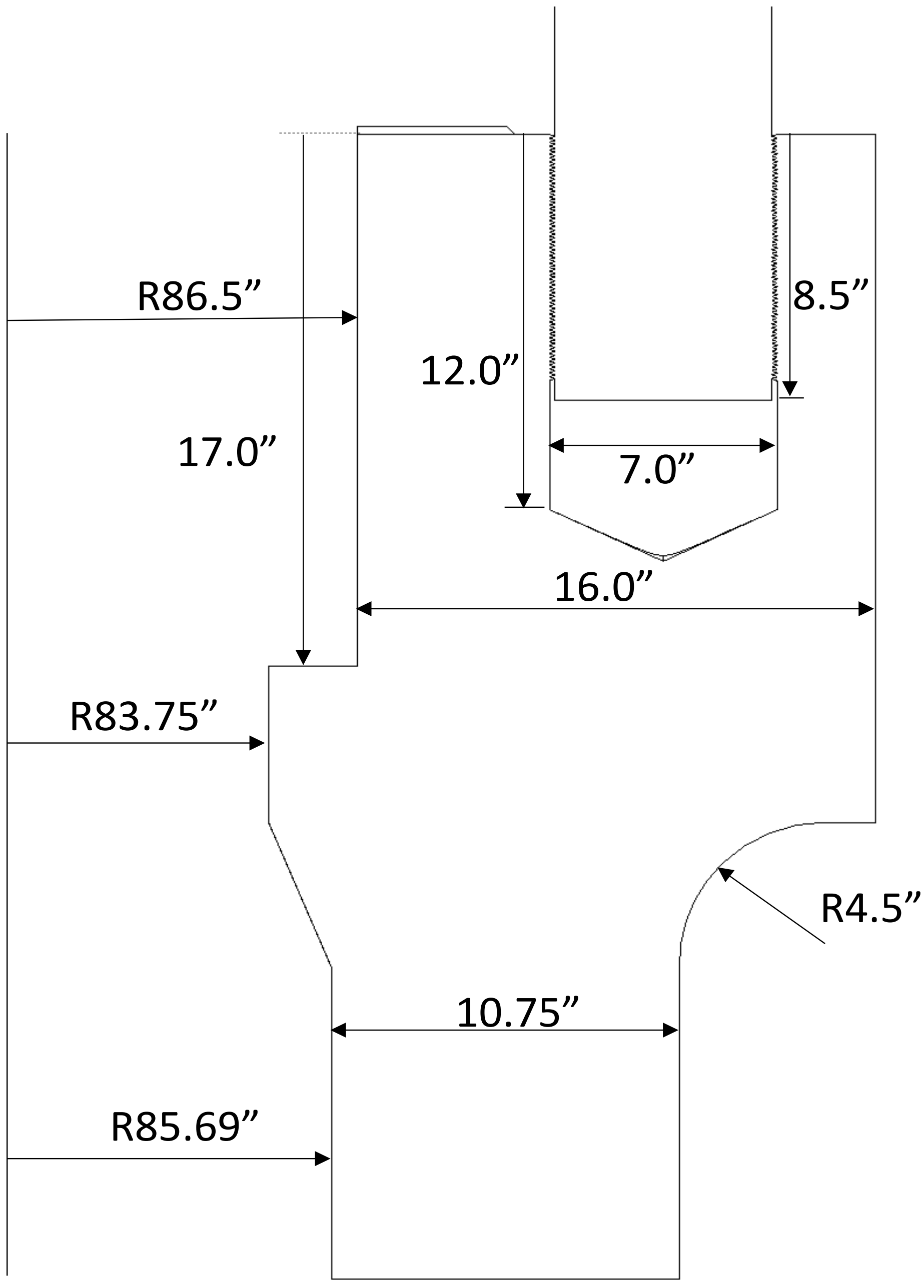


**Figure IR-060-2**
**Finite Element Model Showing Bolt and Flange Connection**

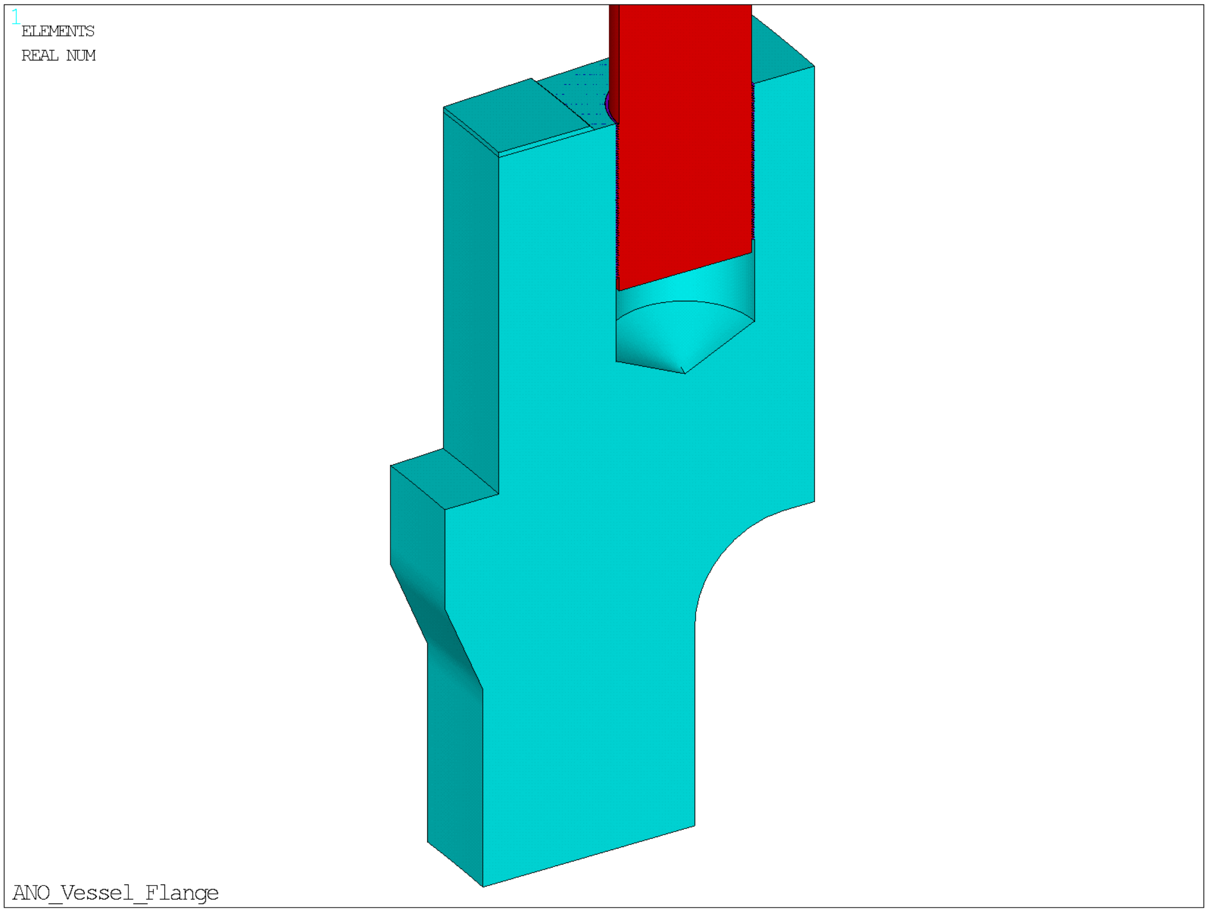


**Figure IR-060-3**
**Finite Element Model Mesh with Detail at Thread Location**

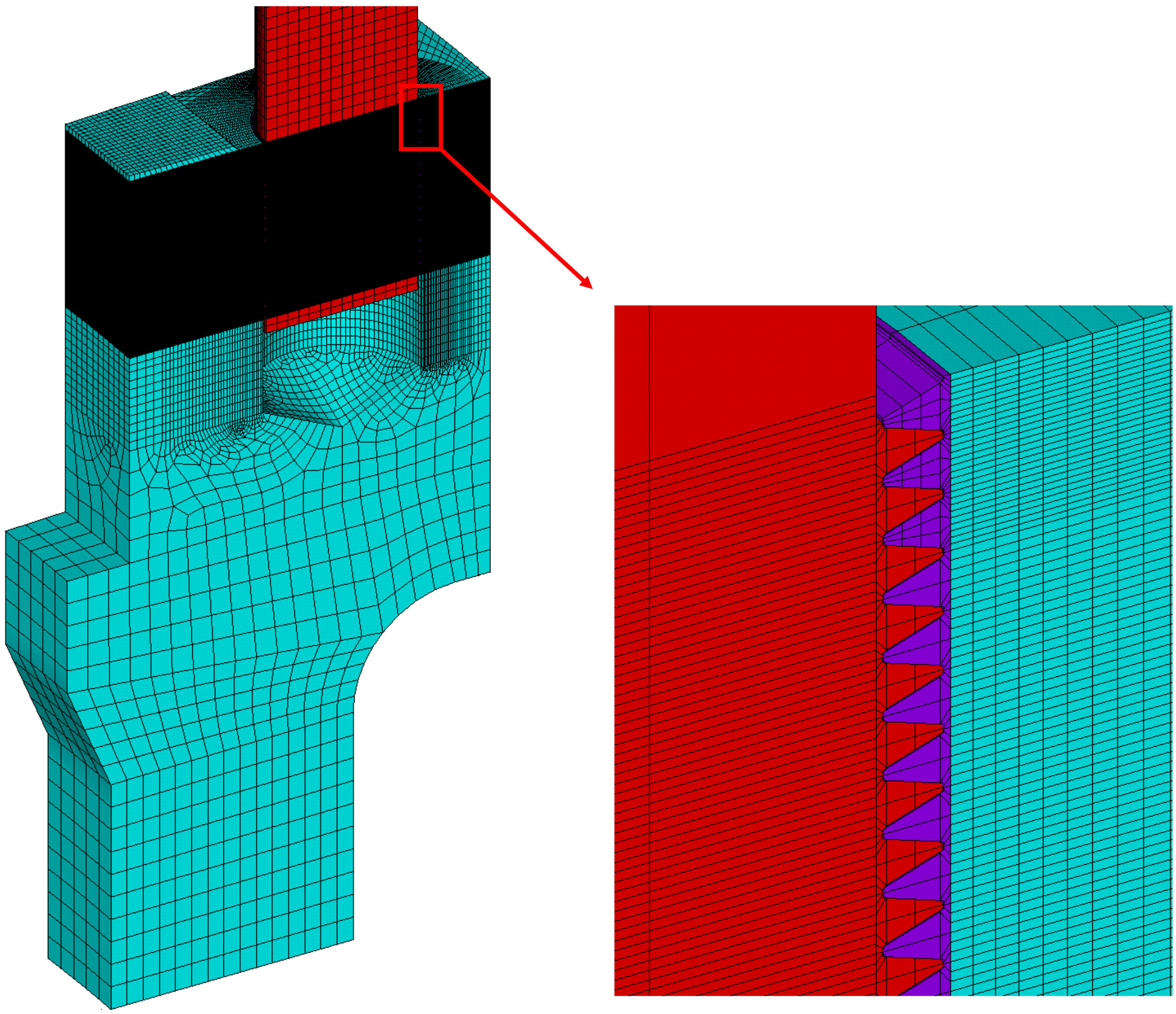


**Figure IR-060-4**
**Cross Section of Circumferential Flaw with Crack Tip Elements Inserted After 10th Thread from Top of Flange**