UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 31, 2012

Mr. Vito A. Kaminskas
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-PY-A290
P.O. Box 97, 10 Center Road
Perry, OH 44081-0097

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)

Dear Mr. Kaminskas:

By letter to the U.S. Nuclear Regulatory Commission (NRC), dated January 24, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML1100320065), as supplemented by letter dated September 9, 2011 (ADAMS Accession No. ML112520658), FirstEnergy Nuclear Operating Company (the licensee), submitted its third 10-year inservice inspection (ISI) interval program plan requests for relief (RRs) IR-001, Revision 3, IR-009, Revision 2, IR-012, Revision 3, IR-013, Revision 2, IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2, for the Perry Nuclear Power Plant, Unit No. 1 (PNPP).

The NRC staff has reviewed the licensee's submittals and concludes that the proposed alternatives contained in RRs IR-027, Revision 2, IR-056, Revision 1, and PT-001, Revision 2, provide an acceptable level of quality and safety. These RRs apply to the third 10-year ISI interval for PNPP, which expires on May 17, 2019. Furthermore, for the proposed alternative contained in RR IR-043, Revision 2, the NRC staff concludes that the licensee has demonstrated that the American Society of Mechanical Engineers Code (ASME Code) examination requirements are a hardship without a compensating increase in quality and safety and the licensee's proposed alternative provides reasonable assurance of that the subject valve bodies in RR IR-043, Revision 2, would maintain their leak tightness even though the ASME Code surface or volumetric examinations, as applicable, are not performed.

For IR-054, Revision 1, the NRC staff has reviewed the submittals regarding the licensee's evaluation of the plant-specific criteria specified in the December 19, 2007, safety evaluation (SE) for the Boiling Water Reactor Vessel and Intervals Project (BWRVIP)-108 report (ADAMS Accession No. ML073600374), which provides the technical bases for use of ASME Code Case N-702, to examine reactor pressure vessel (RPV) nozzle-to-vessel welds and nozzle inner radii at PNPP. The NRC staff determined that the licensee's proposed alternative provides an acceptable level of quality and safety and applies to all PNPP RPV nozzles described in the request, with the exception of feedwater nozzles and control rod drive return nozzles

Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Sections 50.55a(a)(3)(i) and 50.55a(a)(3)(ii), and is in compliance with the requirements of 10 CFR 50.55a with the authorizing of these alternatives contained in RRs IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2, for PNPP. Therefore, the NRC staff authorizes the licensee's proposed alternatives contained in RRs IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2.

In its letter dated April 5, 2011 (ADAMS Accession No. ML111020311), the licensee withdrew its alternatives contained in RRs IR-001, Revision 3, and IR-012, Revision 3. The NRC acknowledged this action in letter dated April 19, 2011 (ADAMS Accession No. ML11050105).

The RRs identified as IR-009, Revision 2, and IR-013, Revision 2, will be handled under separate NRC correspondence.

The NRC staff's SE is enclosed.

Please contact the PNPP Project Manager, Michael Mahoney, at (301) 415-3867 if you have any questions on this action.

Sincerely,

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-440

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ON THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO: STN 50-440

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff, with technical assistance from its contractor, Pacific Northwest National Laboratory (PNNL), has reviewed and evaluated the information provided by FirstEnergy Nuclear Operating Company (the licensee, FENOC), by its letter dated January 24, 2011, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110320065), proposed in its third 10-year inservice inspection (ISI) interval program plan, which expires on May 17, 2019, requests for relief (RRs) for IR-001, Revision 3, IR-009, Revision 2, IR-012, Revision 3, IR-013, Revision 2, IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2, for Perry Nuclear Power Plant, Unit 1 (PNPP). Additionally, in response to a NRC request for additional information, the licensee submitted additional information in its letter dated September 9, 2011 (ADAMS Accession No. ML112520658). In addition, in its letter dated April 5, 2011 (ADAMS Accession No. ML111050105), the licensee withdrew its alternatives contained in RRs IR-001, Revision 3, and IR-012, Revision 3. These alternatives will not be discussed further in this safety evaluation (SE) as well as IR-009, Revision 2 and IR-013, Revision 2, these will be handled under separate correspondence.

The NRC staff adopts the evaluations and recommendations contained in PNNL's Technical Letter Report which has been incorporated into this SE, for authorizing the licensee's alternatives.

2.0 REGULATORY REQUIREMENTS

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

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3, components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ISI ASME Code of Record for the PNPP third 10-year ISI for PNPP is the 2001 Edition through the 2003 Addenda of the ASME Code Section XI. The third 10-year ISI interval for PNPP interval ends on May 17, 2019.

3.0 EVALUATION

The information provided by the licensee in support of the requests for alternatives to ASME Code requirements has been evaluated and the bases for disposition are documented below. For clarity, the licensee's requests have been evaluated in several parts according to ASME Code Examination Category.

3.1 Proposed Alternative IR-027, Revision 2, ASME Code, Section XI, Table IWD-2500-1, Examination Category D-A, Item D1.10, Welded Attachments for Vessels, Piping, Pumps, and Valves

ASME Code Requirement

The ASME Code, Section XI, Table IWD-2500-1, Examination Category D-A, Item D1.10, requires 100-percent visual examination (VT-1), as defined by Figure IWD-2500-1, of the length of the attachment welds for ASME Code, Class 3, pressure vessels.

Licensee's Proposed Alternative to ASME Code

In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the ASME Code-required VT-1 for integrally attached anchor welds 1R45-A003A-WA and 1R45-A003B-WA of Divisions 1 and 2 diesel fuel oil day tanks (day tanks). The licensee's alternative states that a visual examination of the fire retardant coating (Pyrocrete) covering the welded attachments will be performed for conditions which could indicate structural degradation of the attachment welds beneath the coating.

Licensee's Proposed Alternative Examination and Basis for Use: (as stated)

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

> 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 [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 [Code] 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 [ASME] Code examination would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

<u>NRC Staff Evaluation</u>

The ASME Code requires 100-percent VT-1 of Class 3 pressure vessel welded attachments. However, visual examinations of these attachments at PNPP are limited due to a Pyrocrete fire retardant coating. Pyrocrete is considered to be a permanent feature and is intended to endure through the life span of the facility. In order for the licensee to obtain 100-percent of the ASME Code-required examination coverage, the Pyrocrete coating would need to be removed by cutting and chipping and then reapplied to meet fire protection requirements.

Pyrocrete is a cement-like hard coating that is 1-7/16 inches thick on the supports and 3 inches thick in other areas of the day tanks. Removal of this material requires chipping, cutting, or grinding which may cause damage to the subject weld or surrounding areas. The NRC staff determined that for the licensee to remove the Pyrocrete to examine the subject attachments would be a hardship without a compensating increase in quality and safety.

As an alternative, the licensee has proposed that a visual examination of the Pyrocrete covering the welded attachments will be performed to detect conditions which could indicate potential structural degradation of the attachment welds beneath the coating. Since the Pyrocrete forms a rigid bond with the underlying base materials and welds, evidence of any structural damage, should it occur, would be readily detected in the coating.

Based on the visual examination proposed for the Pyrocrete coating adhering to the attachment welds, it is reasonable to conclude that, if significant structural degradation occurs, evidence of it will be detected. The licensee's proposed alternative provides reasonable assurance of structural integrity.

3.2 Proposed Alternative IR-043, Revision 2, ASME Code Section XI, Table IWB-2500-1, Examination Category B-M-1, Items B12.30 and B12.40, Pressure Retaining Welds in Valve Bodies

ASME Code Requirement

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-M-1, Items B12.30 and B12.40, require essentially 100-percent surface or volumetric examination (VT-2), as applicable, as defined by Figure IWB-2500-17, for selected Class 1 valve body welds. "Essentially 100 percent," as clarified by ASME Code Case N-460, is greater than 90-percent coverage of the examination volume, or surface area, as applicable. ASME Code Case N-460, "Alternative Examination Coverage for Class 1 and Class 2 Welds Section XI, Division1," has been approved for use by the NRC in Regulatory Guide (RG) 1.147, Revision 16 (ADAMS Accession No. ML101000536).

Licensee's Proposed Alternative to ASME Code

In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed to perform a VT-2 each refueling outage during the performance of the system leakage test of the ASME Code, Class 1, boundary as an alternative to the ASME Code-required surface and volumetric examinations of ASME Code, Class 1, valve body welds.

Licensee's Proposed Alternative Examination and Basis for Use (as stated)

> Performing surface and volumetric examinations on [ASME Code, Section XI, Table IWB-2500-1,] 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.
>
> In accordance with [ASME Code, Section XI, Table IWB-2500-1,] Examination Category B-P, the welds receive a VT-2 [visual] 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 [Code,] Section III and ASME [Code,] 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 [Code] 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 [ASME Code, Section XI, Table IWB-2500-1] 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 [ASME Code, Section XI] Table IWB-2500-1, [Category B-M-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 Man/Rem (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 [ASME Code, Section XI, Table IWB-2500-1,] 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 [ASME Code, Section XI, Table IWB-2500-1, Category B-M-1 examinations in the ASME Code, Section XI] 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.

NRC Staff Evaluation

The ASME Code requires 100-percent surface examination of Class 1 valve body welds less than nominal pipe size (NPS) 4 and volumetric examination of NPS 4 or larger Class 1 valve body welds. At PNPP, these valves pose a severe radiological hazard to examiners as well as support personnel who remove/reinstall insulation, or build scaffolding. The licensee has estimated personnel exposures ranging from 1.0 to 10.0 Man-Rem (MRem) would be incurred if these examinations are imposed. Requiring the licensee to perform surface or volumetric examination on these valve body welds presents a hardship.

From the technical discussions included in the licensee's submittal, operating experience for the subject carbon steel valve body welds shows no reported weld failures or unacceptable indications having occurred. In addition, under risk-informed assessments, there have been no postulated degradation mechanisms for these welds. If degradation of these valves were to occur, the VT-2 during system pressure tests would be capable of detecting any through-wall leakage. Furthermore, because of the excellent operating history and no known or postulated damage mechanisms, the ASME Code eliminated Category B-M-1 inspections from the 2008 Addenda; this Addenda was recently approved by the NRC in rulemaking issued in July 2011. Considering the above, if the currently-required ASME Code volumetric or surface examinations are imposed on the licensee, no service degradation is likely to be discovered for the subject valve body welds at PNPP. Therefore, the NRC staff determined that the valve bodies would maintain their leak tightness even though the surface and/or volumetric examinations are not performed.

Based on current operating experience that indicates that no service degradation has occurred, along with an absence of any known postulated degradation mechanisms, and considering the excessive personnel radiation exposure that would result from performing the ASME Code-required examinations of the subject valve body welds, along with the fact that later NRC-approved Editions of the ASME Code have eliminated these Category B-M-1 examinations, the NRC staff concluded that performing the current ASME Code-required surface or volumetric examinations due to the high radiation dose of at least 4,600 MRem, which is not in keeping with "As Low As Reasonable Achievable" would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.3 Proposed Alternative IR-054, Revision 1, ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Items B3.90 and B3.100, Full Penetration Welded Nozzles in Vessels

ASME Code Requirement

ASME Code, Section XI, Examination Category B-D, Items B3.90 and B3.100, requires 100-percent volumetric examination, as defined by Figures IWB-2500-7 (a) through (d), as applicable, of all full penetration Class 1 RPV nozzle-to-vessel welds and nozzle inside radius (IR) sections.

Licensee's Proposed Alternative to ASME Code

In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative to ASME Code-required volumetric examinations on ASME Code, Class 1, RPV nozzle-to-vessel welds and nozzle IR sections. The proposed alternative reduces the ASME Code-required 100-percent volumetric examinations of all nozzle-to-shell welds and inner radii, to 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 during each inspection interval. This alternative is contained in ASME Code Case N-702 "Alternative Requirements for Boiling Water Reactors (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1."

Licensee's Proposed Alternative Examination and Basis for Use (as stated)

> In lieu of performing examination on 100 percent of the identified nozzle assemblies, FENOC proposes to perform, in accordance with [ASME] 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.

[ASME] Code Case N-702 proposes that VT-1 examination may be used in lieu of volumetric examination for the inner radii (Item B3.100). The [PNPP] is already using Code Case N-648-1 [*"Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles, Section XI, Division 1"*] in accordance with conditions placed upon the use of [ASME] Code Case N-648-1 by [RG 1.147, Revision 16] which allows VT-1 examination for nozzle [IR]. As [ASME] Code Case N-648-1 is already approved for use at the PNPP, the specific aspect of utilizing VT-1 examinations as allowed by [ASME] Code Case N-702 is not a part of the request. Despite this allowance, volumetric examinations of the nozzle [IR] of the selected recirculation inlet, core spray, low pressure core injection, and jet pump instrumentation nozzles are performed as their nozzle [IR] are not fully accessible from inside the vessel.

Electric Power Research Institute (EPRI) Technical Report 1003557 ["Boiling-Water Reactor Vessel and Internals Project -108 (BWRVIP-108)]: *"Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii"*, provides the technical basis for the use of ASME 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, $< 1\times10^{-6}$ 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) [(ADAMS Accession No. ML073600374)] approving BWRVIP-108 as a basis for using Code Case N-702. Within Section 5 of the NRC SE, it states that each licensee should demonstrate the plant specific applicability of the BWRVIP-108 report to their units in the relief for alternative by meeting the criteria discussed in Section 5 of the NRC SE.

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

The PNPP-specific applicability to each general and nozzle-specific criteria are as follows[1]:

Criterion 1: the maximum RPV heatup/cooldown rate is less than 115 °F/hour
The maximum [RPV] Heatup/Cooldown rate is limited to less than 100°F/hour

Criterion 2: for recirculation inlet nozzles, $(pr/t)/C_{RPV} < 1.15$
$(pr/t)/C_{RPV} = 0.93 < 1.15$

Criterion 3: for recirculation inlet nozzles, $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NOZZLE} < 1.15$
$[p(r_0^2+r^2)/(r_0^2-r^2)]/C_{NOZZLE} = 1.12 < 1.15$

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1 In the September 9, 2011, licensee response to the NRC RAI, values for Criterion 2 and Criterion 4 were modified from the original submittal.

Criterion 4: for recirculation outlet nozzles, $(pr/t)/C_{RPV} < 1.15$
$(pr/t)/C_{RPV} = 1.11 < 1.15$

Criterion 5: for recirculation inlet nozzles, $[p(r_o^2 + r_i^2)/(r_o^2 - r_i^2)]/C_{NOZZLE} < 1.15$
$[p(r_0^2+r_i^2)/(r_0^2-r^2)]/C_{NOZZLE} = 1.03 < 1.15$

NRC Staff Evaluation

The NRC SE dated December 19, 2007 (ADAMS Accession No. ML073600374), on acceptability of BWRVIP-108, specified five plant-specific criteria that licensees must meet in order to demonstrate that BWRVIP-108 results apply to their plants. The five criteria are related to the driving force of the probabilistic fracture mechanics (PFM) analysis for the recirculation inlet and outlet nozzles. It was stated in the NRC SE that the nozzle material fracture toughness-related (RT_{NDT}) values used in the PFM analyses were based on data from the entire fleet of BWR RPVs. Therefore, the BWRVIP-108 PFM analyses are bounding with respect to fracture resistance, and only the driving force of the underlying PFM analyses needs to be evaluated. It was also stated in the NRC SE that except for the RPV heatup/cooldown rate, the plant-specific criteria are for the recirculation inlet and outlet nozzles only because the probabilities of failure, P(FIE)s, for other nozzles are an order of magnitude lower.

The licensee stated that Criterion 1 is satisfied because PNPP maintains a maximum heatup/cooldown rate of 100 °F/hour, well below the 115 °F/hour criterion limit. The licensee addressed in a letter dated September 17, 2008 (ADAMS Accession No. ML082680091), any events during which the heatup/cooldown rate was in excess of 115 °F/hour, however, this is not a concern as Criterion 1 refers only to normal operating conditions, not typical transients.

For the remaining four criteria the licensee provided, in its original and RAI submittal, PNPP's plant-specific data evaluation of the driving force factors, or ratios, against the criteria established in the December 19, 2007, NRC SE. The licensee's calculated results showed that the remaining four criteria are satisfied, and PNNL confirmed the accuracy of the calculations by performing the calculations independently with the provided radius and thickness values.

It should be noted that RPV feedwater nozzles and control rod drive return line nozzles are outside the scope of ASME Code Case N-702 and are, accordingly, outside the scope of this request.

The ASME Code Case N-702 permits a VT-1 of the nozzle inner radius without performing a sensitivity demonstration of detecting a 1-millimeter width wire or crack. This is not consistent with the NRC position established in RG 1.147, Revision 16, regarding ASME Code Case N-648-1, "Alternative Requirements for Inner Radius Examination or Class 1 Reactor Vessel Nozzles, Section XI, Division 1." However, since the licensee's proposed alternative and the basis above stated that they are currently using ASME Code Case N-648-1, subject to the conditions provided in RG 1.147, Revision 16, for examinations of all nozzle inner radii, the inconsistency between ASME Code Case N-702 and the NRC position regarding VT-1 is not an issue in this request. Despite this allowance, volumetric examinations of the nozzle inner radii of 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.

Finally, the licensee indicated that six indications had been detected, four on the low pressure core injection N6A nozzle-to-vessel Weld 1B13-N6A-KA, one on Core Spray N5A nozzle-to-vessel Weld 1B13-N5A-KA, and one on the recirculation inlet N2B nozzle-to-vessel Weld 1B13-N2B-KA. In all cases, the indications were found to be acceptable per ASME Code, Section XI, IWB-3000.

Based on the above evaluation, the licensee meets all five plant-specific criteria specified in the December 19, 2007, NRC SE on the BWRVIP-108 report. This plant-specific evaluation forms the technical basis for accepting the alternative specified in ASME Code Case N-702, therefore, providing an acceptable level of quality and safety.

3.4 Proposed Alternative IR-056, Revision 1, Examination Category B-N-1, Item B13.10, Interior of Reactor Vessel, and Examination Category B-N-2, Item B13.40, Welded Core Support Structures and Interior Attachments to Reactor Vessels

ASME Code Requirement

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-N-1, Item B13.10, and Examination Category B-N-2, Item B13.40, require 100-percent VT-3 examination, on all accessible areas or surfaces, as applicable, of reactor vessel interior and core support structures, respectively.

Licensee's Proposed Alternative to ASME Code

In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed to use the BWRVIP guidelines as an alternative to the ASME Code-required 100-percent VT-3 examinations for the reactor vessel interior and core support structures listed in
Table 3.6.1 below:

Table 3.6.1 – ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-1 and B-N-2

ASME Code Item	RPV Interior and Core Support Structure Components
B13.10	RPV Interior
B13.40	Shroud Support Plate
B13.40	Shroud Support Legs
B13.40	Shroud Horizontal Welds
B13.40	Shroud Vertical Welds
B13.40	Shroud Repairs
B13.40	Top Guide
B13.40	Core support Plate
B13.40	Control Rod Guide Tubes (CRGTs)

Note: In the licensee's response to the NRC RAI, the licensee agreed to withdraw the proposed alternative, BWRVIP-183, "Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines," since the subject guideline is still under review by the NRC staff.

Licensee's Proposed Alternative Examination and Basis for Use: (as stated)

> FENOC requests to use the [BWRVIP] guidelines, endorsed by the [NRC] and implemented by the industry, to perform examinations in accordance with industry initiatives because [ASME] Code inspection requirements have not evolved with BWR inspection experience.
>
> 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. The NRC has agreed with the BWRVIP approach in principle and has issued SEs for these guidelines (References 1-12)[2]. Therefore, use of these guidelines as an alternative to the subject [ASME] Code requirements provide an acceptable level of quality and safety and will not adversely impact the health and safety of the public.
>
> In lieu of the ASME [Code,] Section XI examination requirements, FENOC proposes to perform examinations pursuant to the requirements within the identified BWRVIP guidelines.
>
> The BWRVIP Inspection and Evaluation (I&E) guidelines have recommended aggressive specific inspection by [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 [ASME]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 [ASME] 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"

2 References 1 through 12 provided by the licensee is not included in this report.

- BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate AP 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"

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

Table 1[3] compares current ASME [Code, Section XI, IWB-2500-1,] Examination Category B-N-1 and B-N-2 requirements with the current BWRVIP guideline requirements, as applicable to the [PNPP).] Table 2[4] 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.

The Attachment[5], "Comparison of [ASME] Code Examination Requirements to BWRVIP Examination Requirements," identifies specific examples that compare the inspection requirements of [ASME Code, Section XI,] Table IWB-2500-1, [Examination Categories B-N-1 and B-N-2] 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 ASME Code requirements, provides an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

3 Table 1 provided by the licensee is not included in this report.
4 Table 2 provided by the licensee is not included in this report.
5 Attachment provided by the licensee is not included in this report.

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NRC Staff Evaluation

The ASME Code requires 100-percent visual examination on all accessible areas or surfaces, as applicable, of reactor vessel interior and core support structures, respectively. PNNL has reviewed the licensee's submittal listing the BWRVIP inspection guidelines that have been proposed as alternatives to the ASME Code requirements given above. The NRC has previously reviewed, approved, and issued SEs on all BWRVIP guidelines listed in the licensee's request. It has also been verified that no NRC conditions have been imposed on the use of the BWRVIP guidelines in the approved SEs.

For ASME Code, Section XI, Table IWB-2550-1, Category B-N-1, Item No. B13.10, reactor vessel interior welds, the applicable BWRVIP guidelines require a VT-3 examination on a more frequent basis than that required by the ASME Code, Section XI. For ASME Code, Section XI, Table IWB-2550-1, Category B-N-2, Item B13.40, the applicable BWRVIP guidelines require, as a minimum, the same visual examination method, VT-3, as the ASME Code for integrally welded core support structures, and for specific areas, it requires either an enhanced visual examination technique or ultrasonic. The BWRVIP examination frequency is equivalent or more frequent than the examination required by the ASME Code.

Based on the above evaluation, the BWRVIP guidelines meet or exceed the ASME Code requirements for the examination method and frequencies of the reactor vessel interior and core support structures. For this reason, the proposed alternatives summarized in the licensee's submittal provide an acceptable level of quality and safety.

3.5 Proposed Alternative PT-001, Revision 2, ASME Code, Section XI, Examination Category C-H, Item C7.10, Pressure Retaining Components

ASME Code Requirement

ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H, Item C7.10, requires 100-percent VT-3 examination of all Class 2 pressure retaining components during each inspection period be performed in accordance with system leakage test descriptions found in ASME Code, Section XI, Paragraph IWC-5220. ASME Code, Section XI, Paragraph IWC-5220, states, "The system leakage test shall be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements." In addition, general requirements for system leakage tests are found in ASME Code, Section XI, Paragraph IWA-5000, which for Class 2 insulated components, requires that hold times be applied prior to VT-2 examinations. If the system is not required during normal plant operation, only a 10-minute hold time is necessary. However, for Class 2 systems required to operate during normal plant operation no hold time is required provided the system has been in operation for at least four hours for insulated components and 10 minutes for noninsulated components.

Licensee's Proposed Alternative to ASME Code

In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed to conduct pressure testing of selected small-bore vent, drain, and instrumentation lines in accordance with ASME Code, Class 1 requirements found in ASME Code, Section XI, Paragraphs IWA-5213(a)(1) and IWB-5210, as an alternative to the ASME Code-required Class 2 system leakage tests for insulated components that cannot be isolated from the ASME Code, Class 1, RPV coolant pressure boundary.

The ASME Code, Class 1, requirements do not require a hold time after attaining test pressure and requires a test pressure "not less than the pressure corresponding to 100-percent rated reactor power."

Licensee's Proposed Alternative Examination and Basis for Use: (as stated)

> ASME [Code,] 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 [Code,] Class 2 valves/components cannot be isolated from the reactor coolant pressure boundary (ASME [Code,] Class 1). Conducting the ASME [Code,] Class 2 examinations during the ASME [Code,] Class 1 system leakage test eliminates the hold time with acceptable quality and safety.
>
> In lieu of [ASME Code, Section XI, Paragraphs] IWA-5213(a)(3) and IWC-5210, which requires ASME [Code,] 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 [ASME Code, Section XI, Paragraphs] IWA-5213(a)(1) and IWB-5210, which do not require a hold time.
>
> For those ASME [Code,] Class 2 systems/components attached to the reactor coolant pressure boundary (ASME [Code,] Class 1) that are not provided with either pressure or test isolation, pressure testing would be conducted in accordance with [ASME Code, Section XI, Paragraphs] 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 [Code,] Class 1 boundary) ASME [Code,] Class 2 system valves/components would be examined during the ASME [Code,] 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), [RPV] coolant pressure boundary. The piping systems and their associated components connected to the [RPV] coolant pressure boundary and less than 1 inch in diameter were constructed to the requirements of ASME [Code,] Section III, Subsection NC, and identified as ASME [Code,] Class 2 for in-service inspection. The associated components and component parts are identified by valve number and listed above[6] These piping systems shall be pressurized during the ASME [Code], Class 1 [RPV], coolant pressure boundary system leakage test and a

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6 List printed in licensee submittal and not reproduced in this SE

VT-2 visual examination would be performed. Although the system would not have been in operation for four hours prior to commencing the examinations, the time required to bring the [RPV] coolant system up to test pressure would allow for the detection of leakage.

Within ASME [Code,] 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 [ASME Code, Section XI, Paragraphs] IWA-5213(a)(1) for IWA-5213(a)(3) and the substitution of [ASME Code, Section XI, Paragraphs] IWB-5210 for IWC-5210 satisfies the intent of the [ASME] Code.

NRC Staff Evaluation

The ASME Code requires 100-percent VT-2 examinations be conducted during system pressure tests for all ASME Code, Class 2, pressure retaining components connected to the ASME Code, Class 1, RVP coolant pressure boundary. For ASME Code, Class 2, systems, ASME Code, Section XI, Paragraph IWA-5213(a)(3), requires no holding time, provided the system has been in operation for at least four hours. ASME Code, Section XI, Paragraph IWC-5210, allows system pressure tests to be conducted during system leakage tests for those systems required to operate during normal plant operations.

However, the design of certain ASME Code, Class 2, components prevents these components from being isolated from the ASME Code, Class 1 RPV coolant pressure boundary. As an alternative, the licensee has proposed to use the ASME Code, Class 1, requirements of ASME Code, Section XI, Paragraphs IWA-5213(a)(1) and IWB-5210, in lieu of the ASME Code, Class 2, requirements listed above. ASME Code, Section XI, IWA-5213(a)(1) and IWB-5210, state that a system leakage test requires no holding time after attaining test pressure and temperature, and require tests be conducted at a pressure not less than the pressure corresponding to 100-percent rated reactor power.

The intent of the requirement that the system be in operation for at least four hours prior to VT-2 examinations during an ASME Code, Class 2, system pressure test is to allow any leakage that may exist to penetrate insulation. However, a system leakage test will be conducted on the subject ASME Code, Class 2, components listed in the licensee's submittal are insulated instrumentation, test connection, vent, and drain line/valves, which are not isolable from the ASME Code, Class 1, RPV coolant pressure boundary.

These components will be pressurized during the ASME Code, Class 1, system leakage test, which is required to be conducted on the ASME Code, Class 1, pressure boundary after attaining test pressure and temperature corresponding to 100-percent rated reactor power. The time required bringing the ASME Code, Class 1, system up to pressure and temperature for the ASME Code, Class 1, system leakage test is adequate to detect any leakage from these insulated ASME, Code, Class 2, components. In addition, the licensee will minimize overall personnel radiation exposure by conducting the ASME Code, Class 2, visual examination during the ASME Code, Class 1, system leakage test, eliminating the need to return after an additional 4 hours to conduct the system leakage test. The licensee's alternative to use ASME Code, Class 1, system leakage test conditions for un-isolable portions of these ASME Code, Class 2, vent, drain, and instrumentation lines provides an acceptable level of quality and safety.

4.0 CONCLUSION

The NRC staff has reviewed the licensee's submittals and concludes that the licensee's proposed alternatives contained in RRs IR-027, Revision 2, IR-056, Revision 1, and PT-001, Revision 2, provides an acceptable level of quality and safety. This relief applies to the third 10-year ISI Interval for PNPP, which expires on May 17, 2019. Furthermore, for the proposed alternative contained in RR IR-043, Revision 2, the NRC staff concludes that the licensee has demonstrated that the ASME Code examination requirements are a hardship without a compensating increase in quality and safety and the licensee's proposed alternative provides reasonable assurance of that the subject valve bodies in RR IR-043, Revision 2, would maintain their leak tightness even though the ASME Code surface or volumetric, as applicable, examinations are not performed.

For IR-054, Revision 1, the NRC staff has reviewed the submittal regarding the licensee's evaluation of the plant-specific criteria specified in the December 19, 2007, SE for the BWRVIP-108 report, which provides technical bases for use of ASME Code Case N-702, to examine RPV nozzle-to-vessel welds and nozzle inner radii at PNPP. Based on the evaluation in Section 3.5 of this SE, the NRC staff determined that the licensee's proposed alternative provides an acceptable level of quality and safety and applies to all requested PNPP RPV nozzles, with the exception of feedwater nozzles and control rod drive return nozzles

Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and 10 CFR 50.55a(a)(3)(ii), and is in compliance with the requirements of 10 CFR 50.55a with the authorizing of these alternatives contained in RRs IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2, for PNPP. Therefore, the NRC staff authorizes the licensee's proposed alternatives contained in, IR-009, Revision 2, IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in the subject requests for relief remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: S. Cumbridge, NRR
T. McLellan, NRR

Date of issuance: January 31, 2012

Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Sections 50.55a(a)(3)(i) and 50.55a(a)(3)(ii), and is in compliance with the requirements of 10 CFR 50.55a with the authorizing of these alternatives contained in RRs IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2, for PNPP. Therefore, the NRC staff authorizes the licensee's proposed alternatives contained in RRs IR-027, Revision 2, IR-043, Revision 2, IR-054, Revision 1, IR-056, Revision 1, and PT-001, Revision 2.

In its letter dated April 5, 2011 (ADAMS Accession No. ML111020311), the licensee withdrew its alternatives contained in RRs IR-001, Revision 3, and IR-012, Revision 3. The NRC acknowledged this action in letter dated April 19, 2011 (ADAMS Accession No. ML11050105).

The RRs identified as IR-009, Revision 2, and IR-013, Revision 2, will be handled under separate NRC correspondence.

The NRC staff's SE is enclosed.

Please contact the PNPP Project Manager, Michael Mahoney, at (301) 415-3867 if you have any questions on this action.

Sincerely,

/RA/

Jacob I. Zimmerman, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-440
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