SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
ON THE FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION
REQUEST FOR RELIEF PRR-24
FOR
ENTERGY NUCLEAR OPERATIONS, INC.
PILGRIM NUCLEAR POWER STATION
DOCKET NUMBER: 50-293
(TAC NO. MF4187)

1.0 INTRODUCTION

By letter dated March 12, 2014, (Agencywide Documents Access & Management System (ADAMS) Accession No. ML14077A175), the licensee, Entergy Nuclear Operations, Inc. (Entergy), submitted Request for Relief PRR-24 that addressed alternative examinations to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components* for Pilgrim Nuclear Power Station (PNPS). Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1), the licensee requested to use an alternative to the ASME Code, Section XI inspection requirements regarding examination of certain reactor pressure vessel (RPV) nozzle-to-shell welds and nozzle inner radii at PNPS. The proposed alternative is in accordance with ASME Code Case N-702, *Alternative Requirements for Boiling Water reactors (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds,* without using the visual (VT-1) examination specified in the Code Case. The technical basis for ASME Code Case N-702 was documented in an Electric Power Research Institute (EPRI) report for the Boiling Water Reactor Vessel and Internals Project (BWRVIP), *BWRVIP-241: BWR Vessel Internals Project Probabilistic Fracture Mechanics [(PFM)] Evaluation for the Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii.* The BWRVIP-241 report was approved by the U.S. Nuclear Regulatory Commission (NRC) in a safety evaluation (SE) dated April 19, 2013, which identified plant specific requirements that must be met for applicants proposing to use this alternative.

This request applies to the fourth 10-year inservice inspection (ISI) interval, in which PNPS adopted the 1998 Edition through the 2000 Addenda of ASME Code Section XI as the Code of record. Additionally, in response to an NRC Request for Additional Information (RAI), the licensee submitted further information in a letter dated January 8, 2015 (ADAMS Accession No. ML15016A115). The Staff has evaluated the subject proposed alternative in this SE.

2.0 REGULATORY REQUIREMENTS

The paragraph headings in 10 CFR, Part 50, Section 50.55(a) were changed by *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014 (e.g., 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2)). See the cross-reference tables, which are cited in the notice, at ADAMS Accession No. ML14015A191 and ADAMS package Accession No. ML14211A050.

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.

Inservice inspection of 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 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrate (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The regulations require that ISI 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 Code of record for the fourth 10-year interval at PNPS is the 1998 Edition through the 2000 Addenda of the ASME Code Section XI. The fourth 10-year ISI interval for PNPS is projected to end on June 30, 2015.

The ASME Code, Section XI requires volumetric examination of all (100 percent) RPV nozzle-to-shell welds and nozzle inner radii, during each 10-year ISI interval. ASME Code Case N-702 proposes an alternative which reduces the examinations of RPV nozzle-to-shell welds and nozzle inner radius sections from 100 percent (all nozzles) to 25 percent of the nozzles for each nozzle type during each 10-year interval. The NRC has approved the BWRVIP-241 report, which contains the technical basis supporting ASME Code Case N-702. The April 19, 2013, SE regarding the BWRVIP-241 report provided plant-specific requirements to be satisfied by licensees who propose to use ASME Code Case N-702.

3.0 TECHNICAL EVALUATION

The following plant-specific requirements are those identified by NRC in the April 19, 2013, SE for the BWRVIP-241 report supporting use of the ASME Code Case N-702:

> “However, each licensee should demonstrate the plant specific applicability of the BWRVIP-241 report to their units in the relief request by demonstrating all of the following:
>
> (1) the maximum reactor pressure vessel (RPV) heat-up/cool-down rate is limited to less than 115 ºF/hour;
>
> For recirculation inlet nozzles
>
> (2) $(pr/t)/C_{RPV} < 1.15$

p = RPV normal operating pressure (psi),
r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
C_{RPV} = 19332;

(3) $[p(r_o^2 + r_i^2)/ (r_o^2 - r_i^2)] / C_{NOZZLE} < 1.47$

p = RPV normal operating pressure (psi),
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
C_{NOZZLE} = 1637;

<u>For recirculation outlet nozzles</u>

(4) $(pr/t)/C_{RPV} < 1.15$

p = RPV normal operating pressure (psi),
r = RPV inner radius (inch),
t = RPV wall thickness (inch), and
C_{RPV} = 16171; and

(5) $[p(r_o^2 + r_i^2)/ (r_o^2 - r_i^2)]/C_{NOZZLE} < 1.59$

p = RPV normal operating pressure (psi),
r_o = nozzle outer radius (inch),
r_i = nozzle inner radius (inch), and
C_{NOZZLE} = 1977."

This plant-specific information is required by the NRC staff to ensure that the PFM analysis documented in the BWRVIP-241 report applies to the RPV of the licensee's plant.

3.1 Proposed Alternative PRR-24, ASME Code, Section XI, Examination Category B-D, Items B3.90 and B3.100, Full Penetration Welded Nozzles in Vessels

<u>ASME Code Requirement</u>

ASME Code, Section XI, Examination Category B-D, Items B3.90 and B3.100 require 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-shell welds and nozzle inside radius sections.

<u>Licensee's Proposed Alternative to ASME Code</u>

In accordance with 10 CFR 50.55a(z)(1), the licensee proposed an alternative to ASME Code-required volumetric examinations for the ASME Code, Class 1 RPV nozzle-to-shell welds and nozzle inner radius sections listed below in Table 3.1.1. The proposed alternative reduces the ASME Code-required volumetric examinations for all RPV 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*. The required examination volume for the* *reduced set of nozzles remains at 100 percent of that depicted in Figures IWB-2500-7 (a) through (d), as applicable.*

Table 3.1.1- ASME Code, Section XI, Examination Category B-D

ASME Code Item	Weld ID	Weld Type
B3.90	RPV-N2A-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2B-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2C-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2D-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2E-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2F-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2G-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2H-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2J-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.90	RPV-N2K-NV	12” Recirculation Inlet Nozzle-to-Vessel Weld
B3.100	RPV-N2A-NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2B- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2C- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2D- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2E- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2F- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2G- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2H- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2J- NIR	12” Recirculation Inlet Nozzle Inner Radius
B3.100	RPV-N2K- NIR	12” Recirculation Inlet Nozzle Inner Radius

<u>Licensee's Proposed Alternative Examination and Basis for Use</u>

For three of the Reactor Recirculation nozzle assemblies shown in Table 3.1.2 below, both the inner radius region and the nozzle-to-shell weld have already been examined during the fourth interval.

Table 3.1.2 PNPS Summary – Affected Components of PRR-24				
Group	**Nozzle Description**	**Total Number**	**Minimum Number to be Examined**	**Comments**
N2	Recirculation Inlet	10	3	3 completed in RFO17 (2009)

EPRI Technical Report 1021005, "*BWRVIP-241: BWR Vessel and Internals Project (BWRVIP), Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii*" provides the technical basis for use of Code Case N-702. BWRVIP-241 was developed to propose a relaxation of the criteria in *BWRVIP-108: Boiling Water Reactor Vessel and Internals Project (BWRVIP) Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii* allowing BWR's to obtain inspection relief for their Reactor Recirculation inlet and outlet nozzles. 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 (i.e., $< 1x10^{-6}$ for 40 years) with or without inservice inspection. The report concludes that inspection of 25 percent of each nozzle type is technically justified.

The applicability of the BWRVIP-241 report to PNPS is demonstrated by showing the criteria within Section 5 of the SE are met for the recirculation inlet nozzles and inner radius sections.

The general terms used in the SE Section 5 applicability evaluations are:

$C_{i.RPV}$ = *recirculation inlet nozzles (from BWRVIP-108 model)* = [ ]

$C_{i\text{-}NOZZLE}$ = *recirculation inlet nozzles (from BWRVIP- 108 model)* = [ ]

The PNPS nozzle-specific terms to be used in the SER Section 5 applicability evaluations are as follows:

Heatup / Cooldown rate = [ ]

p = Reactor Pressure Vessel (RPV) normal operating pressure, p = [ ]

r = RPV inner radius, r = [ ]

t = RPV wall thickness, t = [ ]

r_{iN2} = *inner radius for Recirculation Inlet N2 nozzles,* r_{jN2} = [ ]
r_{oN2} - *outer radius for Recirculation Inlet N2 nozzles,* r_{oN2} = [ ]

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 or equal to 100°F/hour

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

$(pr/t)/C_{RPV}$ = [ ]< 1.15

Criterion 3: for recirculation inlet nozzles, $[p(r_o^2+ r_i^{\ 2})/ (r_o^2- r_i^{\ 2})]/C_{NOZZLE} < 1.47$

$[p(r_0^2+r^2)/(r_0^2-r^2)]/C_{NOZZLE}= [\quad] < 1.47$

The results of the equations in Attachment 2 of the licensee's submittal or Criterion 1, 2, and 3 above demonstrate the applicability of the BWRVIP-108 / BWRVIP-241 reports to PNPS by showing the criteria within Section 5.0 of the NRC SE is met for all nozzles listed in Table 3.1.1 above. Therefore, the basis for using Code Case N-702 is demonstrated for the PNPS reactor recirculation inlet nozzles in Table 3.1.1 above.

3.2 <u>Staff Evaluation</u>

An NRC SE dated December 19, 2007, 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 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 heat-up/cool-down 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.

On April 19, 2013, the NRC issued a SE approving the use of BWRVIP-241 which revised Criterion 3 and Criterion 5 that were previously approved in the SE for BWRVIP-108. The BWRVIP performed additional PFM analyses in the BWRVIP-241 report using the bounding recirculation inlet and outlet nozzles instead of the typical recirculation inlet and outlet nozzles of the BWRVIP-108 report. The BWRVIP's additional PFM analyses demonstrated that the limits can be higher than 1.15 and the corresponding probability of failures are still below 5×10^{-6}/yr. Criterion 3 was modified to be less than or equal to 1.47 and Criterion 5 was modified to be less than or equal to 1.59. The NRC found that these changes result in probabilities of failure that are at least two orders of magnitude lower than the NRC safety goal of 5×10^{-6}/yr for the pressurized thermal shock concern. As stated in the NRC SE, the PFM results in BWRVIP-241 are best considered as a supplement to those in BWRVIP-108, not a replacement. However, it should be made clear that the conditions and limitations specified in Section 5.0 of this SE supersede those of the SE for the BWRVIP-108 report.

The licensee stated that Criterion 1 is satisfied because PNPS maintains a maximum heat-up/cool-down rate of 100 °F/hour, well below the 115 °F/hour criterion limit. The licensee stated that in accordance with their Technical Specification 3.6.A.2, Reactor Coolant System heat-up and cool-down rates are limited to a maximum of 100 °F/hour of when averaged over any 1-hour period. This addressed whether there have been any events during which the heat-up/cool-down rate was in excess of 115 of/hour. This is not a concern as Criterion 1 refers only to normal operations, not typical transients.

For Criterion 2 and 3, the licensee provided and confirmed, in its original and RAI submittals, PNPS's plant-specific data evaluation of the driving force factors, or ratios, against the criteria established in the April 19, 2013 NRC SE. The licensee's calculated results showed that Criterion 2 and 3 are satisfied, and the staff confirmed the accuracy of the calculations by performing the calculations independently with the provided radius and thickness values.

Criterion 4 and 5 have been previously approved for the licensee in a NRC SE dated August 25, 2010 for the RPV recirculation outlet nozzles and are acceptable.

The licensee noted that ASME Code Case N-702 stipulates that the VT-1 visual examination method may be used in lieu of the volumetric examination method for the inner radius sections. Despite this allowance, all examination of nozzle inner radii of the selected recirculation inlet nozzles will be volumetric examinations. The licensee has no intention to use Code Case N-648-1 *"Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles Section XI, Division 1."* Finally, the licensee indicated that ten recordable indications had been detected, three on Recirculation Inlet nozzle-to-shell Weld RPV-N2G-NV, four on Recirculation Inlet nozzle-to-shell Weld RPV-N2H-NV, and three on Recirculation Inlet nozzle-to-shell Weld RPV-N2K-NV. 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 Criterion 1, 2, and 3 specified in the April 19, 2013, NRC SE on the BWRVIP-241 report. This plant-specific evaluation forms the technical basis for accepting the alternative specified in ASME Code Case N-702, thus, providing an acceptable level of quality and safety.

4.0 CONCLUSIONS

The Staff has reviewed the licensee's submittals for proposed alternative PRR-24 regarding the evaluation of the plant specific criteria identified in the April 19, 2013, SE for the BWRVIP-241 report, which provides the technical bases for use of ASME Code Case N-702, to examine selected RPV nozzle-to-shell welds and nozzle inner radii at PNPS. Based on the evaluation in Section 3.2 of this report, it has been determined that the licensee's proposed alternative provides an acceptable level of quality and safety, and applies to all subject PNPS RPV nozzles. Therefore the licensee's proposed alternative is authorized pursuant to10 CFR 50.55a(z)(1), for the fourth ISI interval at PNPS.

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.