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

January 5, 2016

Everett Perkins, Jr.
Entergy Nuclear Operations
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER STATION - RELIEF REQUEST PRR-50, USE OF ALTERNATIVES, IMPLEMENTATION OF CODE CASE N-702 (CAC NO. MF6362)

Dear Mr. Perkins:

By letter dated June 4, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15166A037), as supplemented by letter dated October 21, 2015 (ADAMS Accession No. ML15201A255), Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted Request for Relief PRR-50 to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI, "Rules for inservice inspection of Nuclear Power Plant Components," requirements at Pilgrim Nuclear Power Station (PNPS).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 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 licensee will ultrasonically examine the subject welds using ASME Code, Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," as implemented, as required and conditioned by 10 CFR 50.55a(b)(2)(xv). 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." 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 NRC in a safety evaluation (SE) dated April 19, 2013 (ADAMS Accession No. ML13071A240), which identified plant specific requirements that must be met for applicants proposing to use this alternative.

The NRC staff has reviewed the licensee's submittal for its proposed alternative request PRR-50 regarding the evaluation of the plant specific criteria identified in the April 19, 2013, SE for the BWRVIP-241 report, which provides the technical basis for use of ASME Code Case N-702, to examine selected RPV nozzle-to-shell welds and nozzle inner radii at PNPS. The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed SE, that Entergy has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), for the fifth 10-year inservice inspection interval at PNPS.

If you have any questions, please contact Ms. Booma Venkataraman at 301-415-2934 or via e-mail at Booma.Venkataraman@nrc.gov.

Sincerely,

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosure:
Safety Evaluation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF PRR-50

FIFTH 10-YEAR INTERVAL INSERVICE INSPECTION FOR

ENTERGY NUCLEAR OPERATIONS, INC

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

CAC NO. MF6362

1.0 INTRODUCTION

By letter dated June 4, 2015, (Agencywide Documents Access & Management System (ADAMS) Accession No. ML15166A037), as supplemented by letter dated October 21, 2015 (ADAMS Accession No. ML15301A255), Entergy Nuclear Operations, Inc. (Entergy, the licensee), submitted Request for Relief PRR- 50 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 the 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 licensee will ultrasonically examine the subject welds using ASME Code, Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," as implemented, as required and conditioned by 10 CFR 50.55a(b)(2)(xv). 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." 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 (ADAMS Accession No. ML13071A240), which identified plant specific requirements that must be met for applicants proposing to use this alternative.

BWRVIP-108NP, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," specified five plant-

Enclosure

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 a NRC SE dated December 18, 2007 (ADAMS Accession No. ML073600374) 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.

Both the BWRVIP-108NP and BWRVIP-241 reports support ASME Code Case N-702 for reducing the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radius areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. BWRVIP-108NP contains the technical basis supporting ASME Code Case N-702 and BWRVIP-241 provides supplemental analyses for BWR RPV recirculation inlet and outlet nozzle-to-shell welds and nozzle inner radii. BWRVIP-241 was submitted to address the limitations and conditions specified in the December 19, 2007, Safety Evaluation Report for the BWRVIP-108NP report, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii."

This request applies to the fifth 10-year inservice inspection (ISI) interval, in which PNPS adopted the 2007 Edition through the 2008 Addenda of ASME Code Section XI as the Code of record.

The licensee proposed an alternative PRR-26 in its submittal dated November 26, 2014 (ADAMS Accession No ML14342B001) for the fifth 10-year ISI interval and requested an alternative under the provisions of 10 CFR 50.55a(z)(1) to maintain the Non-Destructive Examination, Pressure Testing, and Repair/Replacement Programs to the 2001 Edition through the 2003 Addenda of ASME Section XI except that Appendix VIII of the 2001 Edition with applicable NRC conditions of 10 CFR 50.55a(b)(2)(xv) will be used. The licensee's PRR-26 proposed alternative was approved in the NRC SE dated June 19, 2015 (ADAMS Accession No. ML15166A401) until December 31, 2017, at which time the licensee will update the PNPS fifth 10-year ISI interval program to the 2007 Edition through the 2008 Addenda.

2.0 REGULATORY REQUIREMENTS

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 fifth 10-year interval at PNPS is the 2007 Edition through the 2008 Addenda of the ASME Code Section XI. The fifth 10-year ISI interval for PNPS is projected to end on June 30, 2025.

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.

ASME Code N-702 has been approved for use in Regulatory Guide 1.147, Revision 17 with conditions as noted below:

> The technical basis supporting the implementation of this Code Case is addressed by 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," EPRI Technical Report 1003557, October 2002 (ML023330203) and BWRVIP-241: BWR Vessel and Internals Project, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI Technical Report 1021005, October 2010 (ML11119A041). The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation regarding BWRVIP-108 dated December 18, 2007 (ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 (ML13071A240) are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.

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 is to 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 °F/hour;

For recirculation inlet nozzles

(2) $(pr/t)/C_{RPV} \leq 1.15$

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

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

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

For recirculation outlet nozzles

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

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

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

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- 50, ASME Code, Section XI, Examination Category B-D, Items B3.90 and B3.100, Full Penetration Welded Nozzles in Vessels

ASME Code Requirement

The ASME Code, Section XI, Table IWB-2500-1, 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.

The reactor pressure vessel (RPV) nozzle-to-vessel welds and inner radii subject to this request are listed below in Table 1:

Table 1			
RPV Nozzle-to-Vessel Welds and Inner Radii Subject to this Request			
Identification Number	Description	Total Number	Minimum Number to be examined
N1	Recirculation Outlet	2	1
N2	Recirculation Inlet	10	3
N3	Main Steam Outlet	4	1
N6	Core Spray	2	1
N7	Spare & Abandoned Head Spray	2	1
N8	Head Vent	1	1
N9	Jet Pump Instrumentation	2	1

<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 in Table 1 above. 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 in the ASME Code.

<u>Licensee's Basis for Use</u>

EPRI Technical Report 1021005, "BWRVIP-241: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," and BWRVIP-108 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, "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., $< 1 \times 10^{-6}$ for 40 years) with or without ISI. The report concludes that inspection of 25 percent of each nozzle type is technically justified.

Boiling Water Reactor Vessel Internals Project has issued two topical reports:

> BWRVIP-108NP "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI Technical Report 1003557, October 2002 (ML023330203) and BWRVIP-241 "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to Vessel Shell Welds and Nozzle Blend Radii," EPRI Technical Report 1021005, October 2010 (ML11119A041)

The BWRVIP-108NP report contains the technical basis supporting ASME Boiler and Pressure Vessel Code Case N-702 "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds" for reducing the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radius areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval.

BWRVIP-241 provides supplemental analyses for BWR RPV recirculation inlet and outlet nozzle-to-shell welds and nozzle inner radii. BWRVIP-241 was submitted to address the limitations and conditions specified in the December 19, 2007, Safety Evaluation Report for the BWRVIP-108NP report, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii."

Based on the two evaluations (BWRVIP-241 and BWRVIP-108NP), the failure probabilities due to a low temperature over pressure event at the nozzle blend radius region and the nozzle-to-vessel shell weld for PNPS recirculation inlet and outlet nozzles are very low and meet the NRC safety goal.

Regulatory Guide 1.147, Revision 17 conditionally accepts the use of Code Case N-702 with the following condition "The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation Report regarding BWRVIP-108 dated December 18, 2007 (ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 (ML13071A240) are met."

The analyses for the N2 nozzles in BWRVIP-1 08NP and BWRVIP-241 are based on the assumption that fluence at the nozzles is negligible because the analysis is for the initial 40 years of plant operation and do not address the extended operating period. Based on analysis described in the licensee's letter dated January 24, 2010, (ADAMS Accession No. ML100270054), peak fluence for 54 effective full power years at the N2 nozzles is calculated to be 2.81×10^{17} n/cm2 (which exceeds the fluence limit of 1.0×10^{17} n/cm 2 as contained in 10 CFR 50, Appendix H). Therefore, a plant specific probabilistic fracture mechanics evaluation was performed to supplement the criteria of Code Case N-702 and BWRVIP-241 in order to demonstrate that the probability of failure remains acceptable over this period of extended operation (to 60 years). This analysis was performed using the same methods as were used in BWRVIP-241, with PNPS specific fracture mechanics analyses. The results demonstrate that for the N2 nozzles at PNPS, the probability of failure was less than the NRC safety goal of 5.0×10^{-6} per year. Therefore, the probabilistic fracture mechanics criteria of BWRVIP-241 remain applicable to the PNPS N2 nozzles.

The analyses in BWRVIP-108NP and BWRVIP-241 were based on predicted fatigue crack growth over the initial licensed operating period and assumed additional fatigue cycles in evaluating fatigue crack growth. PNPS is projected to exceed the total number of thermal cycles used in the BWRVIP analysis during the extended operating period. However, the usage factor for the N2 nozzles remains below 1.0. Previous BWRVIP documents have demonstrated that stress corrosion crack (SCC) growth represents the majority of the crack growth and that crack growth due to additional mechanical/thermal fatigue cycles introduced by the extended operation time is insignificant compared to hypothetical SCC growth. Thus, the amount of thermal cycle driven fatigue crack growth due to the extended operation to 60 years is not a controlling factor in the probability of failure of the BWR reactor vessel nozzles.

The average probability of failure is 9.67×10^{-7} per year for the nozzle blend radius, and 1.67×10^{-8}/yr for the nozzle-to-shell weld, both of which are less than the 5.0×10^{-6}/yr criteria.

The examination history of the subject nozzles, excluding the N2 nozzles, was previously provided to the staff by Entergy letter dated July 13, 2010, from Stephen J. Bethay, Director Nuclear Safety Assurance to NRC Document Control Desk (Reference ADAMS Accession No. ML102020257).

The N2 nozzle-to-vessel welds and associated inner radii were volumetrically examined in the first, second and third intervals. Examination history for the fourth interval is dependent on request for alternative, PNPS PRR-024, which at the writing of this request is under review by the staff (Reference PNPS letter dated March 12, 2014 from Joseph R. Lynch, Regulatory Assurance Manager to NRC Document Control Desk (Reference ADAMS Accession No. ML14077A175). Table 2 provides the detailed examination history for the N-2 nozzles.

A plant specific probabilistic fracture mechanics evaluation was performed by Structural Integrity Associates (SIA) to supplement the criteria of Code Case N-702 and BWRVIP-241 to demonstrate that the probability of failure remains acceptable over a period of extended operation to 60 years This document, "Evaluation of Probability of Failure for Recirculation Inlet (N2) in the Nozzle-to-Shell Welds and Nozzle Blend Radii Regions at Pilgrim Nuclear Station," 1400071.301, Revision 0, February 2014 is contained in the licensee's Request for Additional Information (RAI) response dated October 21, 2015.

3.2 Staff Evaluation

The staff reviewed and based its evaluation on the licensee's calculations provided in its submittal dated June 4, 2015, that are identified by NRC in the April 19, 2013, SE for the BWRVIP-241 report supporting use of ASME Code Case N-702, and the SIA report, "Evaluation of Probability of Failure for Recirculation Inlet (N2) in the Nozzle-to-Shell Welds and Nozzle Blend Radii Regions at Pilgrim Nuclear Station."

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. The NRC SE stated 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. The NRC SE also stated 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 x 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 applicability of the BWRVIP-241 report to PNPS is demonstrated by showing the criteria within Section 5 of the NRC SE are met for the recirculation inlet and outlet nozzles, since they are bounding with respect to fracture resistance (although criterion 1 applies to all components). The staff reviewed the licensee's evaluation of the equations with the data provided by the licensee to demonstrate the plant specific applicability of the BWRVIP-241 report to their unit in the proposed alternative contained in Request for Relief PRR-50.

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 Specifications Surveillance Requirement 3.6.A.2, Reactor Coolant System heat-up and cool-down rates are limited to a maximum of 100 °F/hour when averaged over any 1-hour period.

For Criteria 2 and 3, recirculation inlet nozzles (N2), the licensee provided and confirmed in its submittal 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 staff reviewed the licensee's calculated results and found that they showed that Criterion 2 and 3 are satisfied when applying the equations indicated in Section 3.0, "Technical Evaluation," in this SE.

For Criteria 4 and 5, recirculation outlet nozzles (N1), the licensee provided and confirmed in its submittal 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 staff reviewed the licensee's calculated results and found that they showed that Criterion 4 and 5 are satisfied when applying the equations in Section 3.0, "Technical Evaluation," in this SE.

The results of the evaluations in the licensee's submittal for Criteria 1 through 5 demonstrate the applicability of the BWRVIP-108 / BWRVIP-241 reports to PNPS by showing the criteria within Section 5.0 of the NRC SE are met. Since the recirculation inlet and outlet nozzles are limiting, as described above, the basis for using Code Case N-702 is demonstrated for the PNPS RPV nozzle-to-vessel welds and inner radii listed in Table 1 above.

In addition, the licensee provided in its June 4, 2015, submittal a list in Table 2 of the recent volumetric examinations performed on the recirculation inlet (N2) nozzle-to-shell welds and associated inner radius sections. 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. For the other welds listed in Table 2 there were indications found: e.g. slag inclusions, reflectors, geometric reflectors, and inclusions with no change in size. All were acceptable or allowable per the requirements of the ASME Code.

The staff also reviewed the plant specific probabilistic fracture mechanics evaluation for N2 nozzles blend radii and nozzle-to-shell weld for 25 percent ISI for extended operations by SIA contained in the licensee's RAI response dated October 21, 2015. The evaluation was performed by SIA to supplement the criteria of Code Case N-702 and BWRVIP-241 in order to demonstrate that the probability of failure remains acceptable over a period of extended operation of 60 years.

In the SIA report it was determined that the probabilities of failure (PoF) results for the N2 nozzle blend radii and nozzle-to-shell weld for 25 percent ISI for extended operations (with 25 percent inspection for the initial 40 years of operation) would be 9.67×10^{-7} and 1.67×10^{-8} respectively. It also concluded that the PoF per reactor year for the N2 nozzle blend radii and nozzle-to-shell weld at Pilgrim are below the criteria of 5×10^{-6} per year and that N2 nozzles still meet the acceptable failure probability considering the elevated fluence level per ASME Code Case N-702. Therefore, in its review of SIA, "Evaluation of Probability of Failure for Recirculation Inlet (N2) in the Nozzle-to-Shell Welds and Nozzle Blend Radii Regions at Pilgrim Nuclear Station", the staff found the evaluation to be acceptable to support the use of ASME Code Case N-702 and BWRVIP-241 as stated in the licensee's basis for use of its proposed alternative.

4.0 CONCLUSIONS

The Staff has reviewed the licensee's submittal for its proposed alternative contained in PRR-50 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 3.1 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 and inner radii in Table 1 above. Therefore the licensee's proposed alternative is authorized pursuant to10 CFR 50.55a(z)(1), for the fifth 10-year 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.

Principal Contributor: T. Mclellan

Date: 11/12/2015

If you have any questions, please contact Ms. Booma Venkataraman at 301-415-2934 or via e-mail at Booma.Venkataraman@nrc.gov.

Sincerely,

/RA/

Travis L. Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-293

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Safety Evaluation

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