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

April 21, 2015

Mr. John A. Dent, Jr.
Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER PLANT - RELIEF REQUEST PRR-24 REGARDING NOZZLE-TO-VESSEL WELDS AND NOZZLE INNER RADII EXAMINATIONS (TAC NO. MF4187)

Dear Mr. Dent:

By letter dated March 12, 2014, as supplemented by letter dated January 8, 2015, Entergy Nuclear Operations, Inc. (the licensee), submitted Relief Request PRR-24 for authorization of a proposed alternative to certain requirements of the American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code), Section XI, Examination Category B-D for Pilgrim Nuclear Power Station (Pilgrim). Specifically, the licensee proposed to use ASME Code Case N-702 which requires examination of a minimum of 25 percent of the nozzle-to-vessel welds and inner radius sections. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i)[1], the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed PRR-24 and concluded that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff has concluded that the licensee addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, pursuant to 10 CFR 50.55a(z)(1), the NRC staff has authorized the licensee's proposed alternative, as described in PRR-24, for the remainder of Pilgrim's fourth inservice inspection interval, projected to end on June 30, 2015.

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

1 Retitled Section 50.55a(z)(1), as noticed in the *Federal Register* on November 5, 2014 (79 FR 65776).

If you have any questions, please contact the Pilgrim Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

Michael I. Dudek, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST PRR-24

REGARDING NOZZLE-TO-VESSEL WELDS AND NOZZLE INNER RADII EXAMINATIONS

ENTERGY NUCLEAR OPERATIONS, INC.

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated March 12, 2014 (Agencywide Document Access and Management System (ADAMS) Accession No. ML14077A175), as supplemented by letter dated January 8, 2015 (ADAMS Accession No. ML15016A115), Entergy Nuclear Operations, Inc. (the licensee), submitted Relief Request PRR-24 for authorization of a proposed alternative to certain requirements of the American Society of Mechanical Engineer Boiler and Pressure Vessel Code (ASME Code), Section XI, Examination Category B-D for Pilgrim Nuclear Power Station (Pilgrim).

Specifically, the licensee proposed to use ASME Code Case N-702, which requires examination of a minimum of 25 percent of the nozzle-to-vessel welds and inner radius sections. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i)[1], the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

The technical basis for ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactors (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds," was documented in the Electric Power Research Institute (EPRI) Technical Report (TR) 1021005, "BWRVIP-241: BWR Vessel and Internals Project [BWRVIP], Probabilistic Fracture Mechanics [PFM] Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii." In a safety evaluation dated April 19, 2013 (ADAMS Accession No. ML13071A240), the U.S. Nuclear Regulatory Commission (NRC) approved the BWRVIP-241 report, which identified plant specific requirements that must be met for applicants proposing to use this alternative.

1 The paragraph headings in 10 CFR 50.55(a) were changed, as noticed in the *Federal Register* (FR) on November 5, 2014 (79 FR 65776), which became effective on December 5, 2014. See the cross-reference tables, which are cited in the notice and available in ADAMS under Accession Nos. ML14015A191 and ML14211A050.

Enclosure

2.0 REGULATORY EVALUATION

The regulation at 10 CFR 50.55a(g)(4) states, in part, that 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 (ISI) of 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 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 regulations at 10 CFR 50.55a(z) state, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates that (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 and subject to the limitations and modifications listed therein. The applicable ASME Code of record for the fourth 10-year ISI interval at Pilgrim is the 1998 Edition through the 2000 Addenda of the ASME Code Section XI.

For all reactor pressure vessel (RPV) nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI, requires 100 percent inspection during each 10-year ISI interval. However, ASME Code Case N-702 proposes an alternative which reduces 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.

Based on the above, and subject to the technical evaluation which follows, the NRC staff finds that regulatory authority exists to authorize an alternative to items B3.90 and B23.100 of ASME Code, Section XI, Examination Category B-D, as requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Relief Request PRR-24

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 1. The proposed alternative reduces the ASME Code-required volumetric examinations for all RPV nozzle-to-shell welds and inner radii, from a 100 percent 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. The licensee stated that, "the twenty-five

percent sampling level stated in [ASME] Code Case N-702 provides a significant cost savings and reduction in worker dose exposure."

Table 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

For three of the reactor recirculation nozzle assemblies (Group N2, recirculation inlet, total number – 10, minimum number to be examined – 3), both the inner radius region and the nozzle-to-shell weld have already been examined during the fourth interval, three were completed in Refueling Outage 17 in 2009).

> [The] BWRVIP-241 [report] 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 NRC staff's April 19, 2013, safety evaluation states that each licensee should demonstrate the plant-specific applicability of the BWRVIP-241 report to their units in the request for alternative by meeting the criteria discussed in Section 5 of the safety evaluation. As stated in the PRR-24, the applicability of the BWRVIP-241 report is displayed below by demonstrating Pilgrim's conformance with the three criteria applicable to the recirculation inlet nozzles and inner radius sections.

Criterion 1: The maximum RPV heatup/cooldown rate is less than 115 degrees Fahrenheit (°F)/hour

Therefore, Pilgrim having a maximum RPV Heatup/Cooldown rate that is limited to less than or equal to 100 °F/hour < 115 °F/hour.

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

Where:
p = RPV normal operating pressure, p = 1035 pounds per square inch (psi)
r = RPV inner radius, r = 113.40625 inches
t = RPV wall thickness, t = 6.5 inches
C_{RPV} = recirculation inlet nozzles (from BWRVIP-108 model) = 19332 psi

Therefore,
$(p*r/t)/C_{RPV} = 0.93 < 1.15$

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

Where:
r_{iN2} = inner radius for Recirculation Inlet N2 nozzles = 5.75 inches
r_{oN2} = outer radius for Recirculation Inlet N2 nozzles = 9.125 inches
$C_{iNOZZLE}$ = recirculation inlet nozzles (from BWRVIP- 108 model) = 1637 psi

Therefore,
$[p(r_o^2 + r_i^2)/ (r_o^2 - r_i^2)]/C_{NOZZLE} = 1.465 < 1.47$

3.2 NRC Staff Evaluation

By letter dated December 19, 2007 (ADAMS Accession No. ML073600374), the NRC issued a safety evaluation on the acceptability of BWRVIP-108, which 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. In the safety evaluation, it was stated that the nozzle material fracture toughness-related 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 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 for other nozzles are an order of magnitude lower.

The NRC staff's April 19, 2013, safety evaluation, related to the BWRVIP-241 report, revised Criterion 3. 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, the PFM results in the BWRVIP-241 report are best considered as a supplement to those in the BWRVIP-108 report, not a replacement. However, it should be made clear that the conditions and limitations specified in Section 5.0 of the April 19, 2013, safety evaluation supersede those of the December 19, 2007, safety evaluation for the BWRVIP-108 report.

The licensee stated that Criterion 1 is satisfied because Pilgrim 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 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 °F/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 PRR-24 and its January 8, 2015, supplement, Pilgrim's plant-specific data evaluation of the driving force factors, or ratios, against the criteria established in the NRC staff's April 19, 2013, safety evaluation. The licensee's calculated results showed that Criterion 2 and 3 are satisfied, and the NRC staff confirmed the accuracy of the calculations by performing the calculations independently with the provided radius and thickness values.

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 ASME 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 10 recordable indications had been detected, three on the recirculation inlet nozzle-to-shell weld RPV-N2G-NV, four on the recirculation inlet nozzle-to-shell weld RPV-N2H-NV, and three on the 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, as specified in the NRC staff's April 19, 2013, safety evaluation. This plant-specific evaluation forms the technical basis for accepting the proposed alternative specified in ASME Code Case N-702.

4.0 CONCLUSION

As set forth above, the NRC staff has concluded that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

Therefore, pursuant to 10 CFR 50.55a(z)(1), the NRC staff authorizes the licensee's proposed alternative, as described PRR-24, for the remainder of Pilgrim's fourth ISI interval, projected to end on June 30, 2015.

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

Principal Contributor: T. McLellan

Date: April 21, 2015

If you have any questions, please contact the Pilgrim Project Manager, Nadiyah Morgan, at (301) 415-1016.

Sincerely,

/RA/

Michael I. Dudek, Acting 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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