Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

June 4, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Request for Approval of Pilgrim Relief Requests (PRR)-50 and -51, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of Code Case N-702 (PRR-50) and Implementation of BWRVIP-05 (GL 98-05) (PRR-51)

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No.: 50-293
License No.: DPR-35

LETTER NUMBER: 2.15.032

Dear Sir or Madam:

Pursuant to 10 Code of Federal Regulations (CFR) 50.55a(z)(1), Pilgrim Nuclear Power Station (PNPS) hereby requests an alternative to specific portions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components," on the basis that the proposed alternatives provide an acceptable level of quality and safety. The requests are summarized below. More specific details are provided in Attachments 1 and 2, respectively.

PRR-50

ASME Section XI, 2007 Edition through the 2008 Addenda, Table IWB-2500-1, Examination Category B-D, "Full Penetration Welded Nozzles in Vessels" requires a volumetric examination of all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles each 10-year interval. Additionally, for ultrasonic examinations, ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," is implemented, as required and conditioned by 10 CFR 50.55a(b)(2)(xv). The reactor pressure vessel nozzle-to-vessel welds and inner radii subject to this request are listed below in Table 1 of Attachment 1.

Code Case N-702 was incorporated into Regulatory Guide (RG) 1.147, Revision 17. As stated in the RG, applicability of the Code Case must be shown by demonstrating that the appropriate sections of the U.S. Nuclear Regulatory Commission (NRC) Safety Evaluation Reports for BWRVIP-108 or BWRVIP-241 are met and the evaluation demonstrating the applicability of the Code Case is reviewed and approved by the NRC prior to the application of the Code Case. Accordingly, PNPS is submitting this request to show compliance to the conditions imposed on BWRVIP-241 and to address operating conditions not bound by BWRVIP-108 (fluence and cycles).

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NRR

Pursuant to 10 CFR 50.55a(z)(1), PNPS requests approval to implement the alternative of Code Case N-702 in lieu of the code required 100% examination of all nozzles identified in Table 1. See Attachment 1 for additional details.

PRR-51

ASME Section XI, 2007 Edition through the 2008 Addenda, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, requires a volumetric examination of the circumferential shell welds each interval. PNPS is requesting an alternative in accordance with 10 CFR 50.55a(z)(1) on the basis that this alternative provides an acceptable level of quality and safety. This request for alternative would provide relief from circumferential weld examinations required by the ASME Section XI Code for the extended period of operation.

PNPS is currently in the fourth 10-year In-Service Inspection (ISI) interval which began on July 1, 2005 and ends on June 30, 2015. The ISI Code of Record for the fourth interval is ASME Section XI, 1998 edition through 2000 Addenda.

PNPS requests NRC Staff review and approval of this Request for Alternative as soon as possible following the start of the PNPS fifth 10-year ISI interval which is scheduled to begin on July 1, 2015.

There are no new commitments included in this document.

If you have questions concerning this letter, please contact me at 508-830-8323.

Sincerely,

Everett P. Perkins Jr.

Everett (Chip) P. Perkins Jr.
Manager, Regulatory Assurance

EPP/pm

Attachments:

1. Pilgrim Nuclear Power Station 10 CFR 50.55a Request No. PRR-50, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of Code Case N-702 (7 Pages)

2. Pilgrim Nuclear Power Station 10 CFR 50.55a Request No. PRR-51, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of BWRVIP-05 (GL 98-05) (5 Pages)

cc: Mr. Daniel H. Dorman
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USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

ATTACHMENT 1

TO

LETTER 2.15.032

Pilgrim Nuclear Power Station 10 CFR 50.55a Request No. PRR-50, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of Code Case N-702

(7 Pages)

1. American Society of Mechanical Engineers (ASME) Code Component(s) Affected

Code Class:	ASME Section XI Code Class 1
Component Numbers:	Various (see Table 1 for detailed list of components)
Code References:	ASME Section XI, 2007 Edition with 2008 Addenda Code Case N-702
Examination Category:	B-D
Item Number(s):	B3.90 and B3.100
Unit/Inspection Interval	Pilgrim Nuclear Power Station / Fifth 10-year interval July 1, 2015 – June 30, 2025

2. Applicable ASME Code Requirements

ASME Section XI, 2007 Edition through the 2008 Addenda, Table IWB-2500-1, Examination Category B-D, "Full Penetration Welded Nozzles in Vessels" requires a volumetric examination of all nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles each 10-year interval. Additionally, for ultrasonic examinations, ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," is implemented, as required and conditioned by 10 Code of Federal Regulations (CFR) 50.55a(b)(2)(xv)[1].

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

3. Reason for Request

U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.147 Rev. 17 conditionally accepts the use of Code Case N-702. This code case provides an alternative to performing examination of 100% of the Nozzle-to-Vessels Welds and Inner Radii for Examination Category B-D nozzles with the exception of the Feedwater and Control Rod

[1] Proposed Alternative PRR-26 (submitted under separate cover) is requesting 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.

Drive Return Line (CRDRL) Nozzles. The alternative is to perform examination of a minimum of 25% of the nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size, excluding the Feedwater and CRDRL Nozzles. As stated in the RG, the evaluation demonstrating the applicability of the Code Case must be reviewed and approved by the NRC prior to the application of the Code Case.

4. Proposed Alternative and Basis for Use

Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), Pilgrim Nuclear Power Station (PNPS) requests approval to implement the alternative of Code Case N-702 in lieu of the code required 100% examination of all nozzles identified in Table 1. As an alternative, for the nozzle-to-shell welds and inner radii identified in Table 1, PNPS proposes to examine a minimum of 25% of the nozzle-to-vessel welds and inner radius sections, including at least one nozzle from each system and nominal pipe size, in accordance with Code Case N-702.

Basis for Use

Boiling Water Reactor (BWR) Vessel Internals Project (BWRVIP) 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 (ML-023330203) and
- BWRVIP-241 "Probabilistic Fracture Mechanics Evaluation for the Boling 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 Reportfor 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 (LTOP) 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.

RG 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."

Section 5.0 of the NRC Safety Evaluation for BWRVIP-241 states:

"Licensees who plan to request relief from the ASME Code Section XI requirements for RPV nozzle-to-vessel shell welds and nozzle inner radius sections may reference the BWRVIP-241 report as the technical basis for use of ASME Code Case N-702 as an alternative. 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 RPV heatup/cooldown rate is limited to less than 115°F/hour"

PNPS Technical Specification 3.6.A.2, Reactor Coolant System heatup and cooldown rates are limited to a maximum of 100°F when averaged over any one hour period and thus meets the requirement of condition 1.

Note: Inputs used in 2 through 5 representing the PNPS configuration are in bold text.

Recirculation inlet nozzles (N2)[2]

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

p = RPV normal operating pressure (psi) **(1035)**
r = RPV inner radius (inch) **(113.4063)**
t = RPV wall thickness (inch), **(6.5)** and
C_{RPV} – 19332;

PNPS specific calculations for Condition 2 above:

(1035 x 113.4063)/6.5)/19332 = 0.93 $\leq$ 1.15

The PNPS result is 0.93 and thus meets the requirement of condition 2 to be $\leq$ 1.15.

(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) **(1035)**
r_o = nozzle outer radius (inch) **(9.125)**
r_i = nozzle inner radius (inch), **(5.750)** and
C_{NOZZLE} = 1637;

PNPS specific calculations for Condition 3 above:

$[1035(9.125^2 + 5.75^2)/(9.125^2 - 5.75^2)]/1637 = 1.465 \leq 1.47$

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[2] There is an inconsistency in reported values for the PNPS RPV dimensions in industry documents. Report BWRVIP-241, page 4-1 Section 4.1 and Figure 4-1, on page 4/12 lists the wall thickness as 7 inches which is a typographical error. The value provided in BWRVIP-108 of 5.531 inches was taken from the upper and lower intermediate shell course values, when it should have been the lower shell course. The values used in the BWRVIP-241 calculations and in this document are based on the correct dimensional information of 6.5 inches for RPV wall thickness and 113.4063 inches for RPV inner radius.

The PNPS result is 1.465 and thus meets the requirement of condition 3 to be $\leq$ 1.47

Recirculation outlet nozzles (N1)

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

p = RPV normal operating pressure (psi) **(1035)**
r = RPV inner radius (inch) **(113.4063)**
t = RPV wall thickness (inch), **(6.5)** and
C_{RPV} = 16171; and

PNPS specific calculations for Condition 4 above:

(1035 x 113.40625)/6.5)/16171 = 1.12 $\leq$ 1.15

The PNPS result is 1.12 and thus meets the requirements of condition 4 to be $\leq$ 1.15

(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) **(1035)**
r_o = nozzle outer radius (inch) **(22.3125)**
r_i = nozzle inner radius (inch), **(13.031)** and
C_{NOZZLE} = 1977.

PNPS specific calculation for Condition 5 above:

$[1035(22.3125^2 + 13.031^2)/(22.3125^2 - 13.031^2)]/1977 = 1.06 \leq 1.59$

The PNPS result is 1.06 and thus meets the requirements of condition 5 to be $\leq$ 1.59

The analyses for the N2 nozzles in BWRVIP-108NP 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 PNPS letter 2.10.005, Kevin Bronson, Pilgrim Site Vice President to NRC Document Control Desk, January 24, 2010, peak fluence for 54 effective full power years (EFPY) at the N2 nozzles is calculated to be 2.81×10^{17} n/cm^2 (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[3] 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

[3] Structural Integrity Associates, "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

PNPS, the probability of failure was less than the NRC safety goal[4] 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} per year for the nozzle-to-shell weld, both of which are less than the 5.0×10^{-6} per year 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.

5. Duration of Proposed Alternative

The duration of this request is for the fifth 10-year In-Service Inspection interval ending June 30, 2025.

6. Precedents

Relief PNPS PRR-20 was approved for the fourth 10-year interval as documented in NRC Safety Evaluation Report dated August 25, 2010 (Accession No. ML102290163). This relief excluded the N2 nozzles which have been subsequently addressed by relief PNPS PRR-24 and Safety Evaluation Report dated April 21, 2015 (Accession No. ML15103A069).

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[4] Technical Basis for Revision of Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61), NUREG-1806, Vol. 1, August 2007

TABLE 2					
Inspection History for the Recirculation Inlet Nozzles					
Component ID	Description	Code Category	Code Item	Inspections	Indications
RPV-N2A-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1977, 1984, 1999, 2009	1977 (note 1) 1984 (note 6)
RPV-N2A-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1977, 1984, 1999, 2009	NRI
RPV-N2B-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1980, 1984, 1999, 2009	1984 (note 7)
RPV-N2B-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1980, 1984, 1999, 2009	NRI
RPV-N2C-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1974, 1984, 1999, 2009	NRI
RPV-N2C-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1974, 1984, 1999, 2009	NRI
RPV-N2D-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1974, 1995, 2003	NRI
RPV-N2D-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1974, 1995, 2003	NRI
RPV-N2E-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1980, 1985, 2003	1980 (note 3)
RPV-N2E-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1980, 1985, 2003	NRI
RPV-N2F-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1977, 1982, 1995, 2003	1977 (note 2)
RPV-N2F-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1977, 1982, 1995, 2003	NRI
RPV-N2G-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1977, 1993, 2005	2005 (note 8)
RPV-N2G-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1977, 1993, 2005	NRI
RPV-N2H-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1980, 1993, 2005	1980 (note 4) 2005 (note 9)
RPV-N2H-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1980, 1993, 2005	NRI
RPV-N2K-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1981, 1993, 2005	2005 (note 10)

TABLE 2 (Continued)					
Inspection History for the Recirculation Inlet Nozzles					
Component ID	Description	Code Category	Code Item	Inspections	Indications
RPV-N2K-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1981, 1993, 2005	NRI
RPV-N2J-NV	12" Recirc Inlet Nozzle to Vessel Weld	B-D	B3.90	1980, 1995, 2003	1980 (note 5)
RPV-N2J-NIR	12" Recirc Inlet Nozzle to Inner Radius	B-D	B3.100	1980, 1995, 2003	NRI

Notes:

1. N2A (NV) – 1977, Code acceptable slag inclusions – acceptable
2. N2F (NV) – 1977, Code acceptable reflectors were observed – acceptable
3. N2E (NV) – 1980, Geometric reflectors – acceptable
4. N2H (NV) – 1980, Indications are geometric reflectors from the nozzle ID surface – acceptable
5. N2J (NV) – 1980, Indications are geometric reflectors from the nozzle ID surface – acceptable
6. N2A (NV) – 1984, Reflectors were seen, same inclusion as 1977 no change in size – acceptable
7. N2B (NV) – 1984, Indications found but are allowable and not recordable – acceptable
8. N2G (NV) – 2005, (3) indications, were acceptable per requirements of ASME Section XI
9. N2H (NV) – 2005, (4) indications were acceptable per requirements of ASME Section XI
10. N2K (NV) – 2005, (3) indications were acceptable per requirements of ASME Section XI

ATTACHMENT 2

TO

LETTER 2.15.032

Pilgrim Nuclear Power Station 10 CFR 50.55a Request No. PRR-51, Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1), Implementation of BWRVIP-05 (GL 98-05)

(5 Pages)

1. American Society of Mechanical Engineers (ASME) Code Component(s) Affected

Code Class:	ASME Section XI Code Class 1
Component Numbers:	RPV Circumferential Shell Welds (RPV-C-1-344, RPV-C-9-338, RPV-C-3-339A, and RPV-C-3-339B)
Code References:	ASME Section XI, 2007 Edition with 2008 Addenda
Examination Category:	B-A
Item Number(s):	B1.11
Unit/Inspection Interval:	Pilgrim Nuclear Power Station / Fifth 10-year interval July 1, 2015 – June 30, 2025

2. Applicable ASME Code Requirements

ASME Section XI, 2007 Edition through the 2008 Addenda, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, requires a volumetric examination of the circumferential shell welds each interval.

3. Reason for Request

During the U.S. Nuclear Regulatory Commission (NRC) staff's review of the Pilgrim Nuclear Power Station (PNPS) License Renewal Application (LRA), Request for Additional Information (RAI) 4.2.5-2 dated September 8, 2006 asked PNPS whether it intended to apply for relief from the ASME Code reactor vessel (RV) circumferential weld examination requirements for the extended licensed period.

The PNPS response dated October 8, 2006 indicated that a request for alternative under the provisions of 10 Code of Federal Regulations (CFR) 50.55a would be submitted to exclude the reactor pressure vessel (RPV) shell circumferential welds from examination. Based on PNPS' response, the RAI was considered resolved.

As such, PNPS is requesting an alternative in accordance with 10 CFR 50.55a(z)(1) on the basis that this alternative provides an acceptable level of quality and safety. This request for alternative would provide relief from circumferential weld examinations required by the ASME Section XI Code for the extended period of operation.

PNPS was previously granted this relief for the remainder of the original 40-year license term (Reference TAC No. MB6074, Accession No. ML030640204).

4. Proposed Alternative and Basis for Use

Proposed Alternative

PNPS requests to use BWRVIP-05, with supporting information described herein as the bases for excluding the RPV shell circumferential welds from the examination required by ASME Section XI, Examination Category B-A, Item No. B1.11 for the extended license period ending on June 8, 2032.

The axial weld seams (Examination Category B-A, Item No. B1.12) and their intersection with the associated circumferential weld seams will be examined in accordance with ASME Section XI except where specific relief is granted when essentially 100% (>90%) coverage cannot be obtained.

Basis for Use

The technical basis supporting the requested alternative is provided by BWRVIP-05, (EPRI TR-105697) "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" as accepted in the staff's final safety evaluation report enclosed in a July 28, 1998, letter from Mr. G.C. Lanais, NRC, to Mr. C. Terry, the BWRVIP Chairman. In this letter, the staff concluded that because the failure frequency for circumferential welds in Boiling Water Reactor (BWR) plants is significantly below the criterion specified in Regulatory Guide (RG) 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and below the core damage frequency of any BWR plant, continued inspection would result in a negligible decrease in an already acceptably low RV failure probability and justify elimination of the In-Service Inspection requirements for RV circumferential welds. The staff's letter indicated that BWR applicants may request relief from ASME Code Section XI requirements for volumetric examination of circumferential RV welds by demonstrating that (1) at the expiration of the license the circumferential welds will satisfy the staff's July 28, 1998, evaluation of the limiting conditional failure probability for circumferential welds and (2) the applicants have implemented operator training and established procedures that limit the cold over-pressure event frequency to that specified in the staff's Safety Evaluation Report. The letter also indicated that the requirements for inspection of RV circumferential welds during an additional 20-year license renewal period would need plant-specific reassessment as part of any BWR LRA. The applicant also must request relief from the ASME Code Section XI requirements for volumetric examination of circumferential welds for the extended license term in accordance with 10 CFR 50.55a(g).

Additional information to support the proposed alternative for the license renewal period is contained in the PNPS LRA and its amendments as accepted by the staff in its Safety Evaluation Report dated June 28, 2007. The one exception requiring further action by PNPS affecting the RV Fluence Time-Limited Aging Analyses (TLAAs) was License Renewal Condition (LRC) 4.2.6 which has been closed, as described below.

During the staff's review of the PNPS LRA, the staff found that the RV Fluence TLAA evaluation in Section 4.2.1 of the original LRA unacceptable due to lack of benchmarking data in support of the plant-specific fluence calculations. This was identified as Open Item (OI) 4.2 which became Commitment 47.

To close the OI, PNPS proposed an alternative analysis to address all fluence related TLAAs for the extended operating period.

The alternative analysis assumed increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper-shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria. This

analysis also shows that there is a large margin available to this limiting fluence at the end of the period of extended operation.

The analysis determined that the limiting fluence value was set by a maximum mean RT_{NDT} value for the axial weld failure probability of 114 EF, in order for the axial weld failure frequency to remain below 5 x 10^{-6} per reactor operating year. The corresponding maximum allowable inner diameter (ID) fluence for the RV axial welds was determined to be 3.37 x 10^{18} n/cm^2. If the fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for all fluence-related TLAAs. To confirm that the limiting fluence will not be reached during the period of extended operation and consequently that all of the fluence-related TLAAs remain valid, Commitment 48, was added, but subsequently superseded by LRC 4.2.6.

"The staff issued License Condition 4.2.6: *On or before June 8, 2010, the applicant (Entergy) will submit to the NRC correctly benchmarked RV neutron fluence calculations, consistent with RG 1.190, that will confirm that the neutron fluence for the lower intermediate shell axial welds, at the inner surface of the RV, will not reach the limiting value of 3.37 x 10^{18} n/cm^2 (E>1.0 MeV) by the end of the period of extended operation (54 EFPY)."*

LRC 4.2.6 was addressed by PNPS letter dated January 24, 2010 "Proposed License Amendment to Technical Specifications: P-T Limit Curves and Relocation of Pressure-Temperatures (P-T) Curves to the Pressure and Temperature Limits Report (PTLR)." In part, information provided to the staff in response to LRC 4.2.6 stated:

> *Pilgrim has been a participant in the NRC approved BWRVIP Integrated Surveillance Program as authorized by License Amendment No. 209. As such, Pilgrim opted to use the Monticello Nuclear Power Plant, a BWR/3 class plant, benchmarking evaluation to produce benchmarked Pilgrim-specific fluence and ART values, and revised P-T curve. Entergy has determined that Monticello reactor pressure vessel fluence calculation for a BWR/3 provides an acceptable benchmark for Pilgrim fluence data to support revised P-T Curves for Pilgrim Operating Cycle 18 and beyond. This information was discussed with the NRC Staff on or about October 17, 2008. The NRC staff concurred with the Entergy approach to use Monticello fluence for benchmarking Pilgrim RAMA fluence calculation (as documented in NRC ADAMS Accession Number ML090370920) and to submit Pilgrim revised P-T curves for NRC approval.*

In an NRC Inspection Report dated June 13, 2012, an inspection was performed to review license renewal activities under Temporary Instruction (TI) 2516/001, "Review of License Renewal Activities." The inspection reviewed the completion of commitments made during the renewed license application and compliance with the Commission's rules and regulations and the conditions of PNPS operating license. For LRC 4.2.6 (Commitment 48) the following was documented in the Inspection Report:

> *Commitment 48 provides that on or before June 8, 2010, Entergy will submit to the NRC calculations consistent with Regulatory Guide 1.190 that will demonstrate limiting fluence values will not be reached during the period of extended operation.*
>
> *By way of Entergy's January 24, 2010,"Proposed License Amendment to Technical Specifications: Revised P-T Limit Curves and Relocation of Pressure-Temperature*

(P-T) Curve to the Pressure and Temperature Limits Report (PTLR)," Entergy submitted calculations consistent with Regulatory Guide 1.190 that demonstrated limiting fluence values will not be reached during the period of extended operation.

The inspectors determined that Entergy had implemented Commitment 47.

Note that Commitment 47 stated, "On or before September 15, 2007, submit to the NRC an action plan to improve benchmarking data to support approval of new P-T curves for Pilgrim." Accordingly, the Commitment referenced in the quoted sentence above should have been Commitment 48.

PNPS considers LRC 4.2.6 closed, thus allowing use of the fluence values calculated by the methods described in the LRA and the staff's Safety Evaluation Report.

In accordance with the guidance of Generic Letter 98-05, "Boiling Water Reactor Licensees Use of BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," applicants requesting relief from examination of the subject welds are to demonstrate that: (1) at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998 safety evaluation, and (2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998 safety evaluation.

(1) Satisfying the Limiting Conditional Failure Probability for Circumferential Welds

Section 4.2.5.2 of the Safety Evaluation Report for the LRA states in part:

"...the staff found that the applicant correctly applied the 64 EFPY mean RT_{NDT} value of 128.5 °F from Table 2.6-5 of the staff SER on the BWRVIP-05 Report in the back-calculation of the maximum allowable 54 EFPY fluence on this TLAA. The staff used this mean RT_{NDT} value in its evaluation of the BWRVIP-05 Report for determining an acceptable circumferential weld conditional failure probability. The 128.5 °F 64 EFPY mean RT_{NDT} value from the staff SER on the BWRVIP-05 Report is characteristic of welds by Combustion Engineering, which fabricated the circumferential welds in the RV."

The staff found the TLAA for the RV circumferential weld examination relief acceptable in accordance with 10 CFR 54.21(c)(1)(ii). As reported in SER 4.2, the staff has reviewed and accepted Commitment 48 (replaced by License Condition 4.2.6) and the alternative fluence analysis for resolving OI 4.2 and confirming the results of the TLAAs.

Even though the initial PNPS fluence calculations were performed prior to the Monticello benchmarking, there have been no material changes in the RAMA code and methodology that would have impacted the results of fluence and Adjusted Reference Temperature (ART) calculations. Therefore the unbiased fluence and ART values included in TransWare Report ENT-FLU-001-R-001, Rev. 0 provides predictable PNPS reactor vessel fluence values using RAMA methodology.

The PNPS fluence calculation predicts a peak fluence for 54 effective full power years (EFPY) of 1.14×10^{18} n/cm^2 at axial welds 1-338A, B, and C in the intermediate shell and 1.28×10^{18} n/cm^2 at the lower intermediate shell location. The peak fluence at the

bounding circumferential weld is 8.69×10^{17} n/cm^2. See Table 1 below. Table 2 shows the most limiting weld which bounds the circumferential weld at 54 EFPY.

TABLE 1						
Bounding Circumferential Weld						
Location[1]	Wall Thickness (inches)		Fluence at ID	Attenuation, 1/4t	Fluence at 1/4t	Fluence Factor, FF
	Full[2]	1/4t	(n/cm^2)	$e^{-0.24x}$	(n/cm^2)	$f^{(0..28-0.1logf)}$
L. int Shell Girth Weld 1-344	5.531	1.383	8.69×10^{17}	0.718	6.24×10^{17}	0.329

Note 1: Fluence values are from TransWare Report No. ENT-FLU-R-001, Rev. 0, "Pilgrim Nuclear Power Station Reactor Pressure Vessel Fluence Evaluation", Tables 7-3 and 7-4.

Note 2: RPV minimum thickness is 5 17/32 inches per Section 3.3.2 of SIR-00-082, Revision 0.

TABLE 2											
Structural Integrity Report No. PNPS-22Q-301, "RT_{NDT} and ART Evaluation" information for the bounding circumferential weld at 54 EFPY											
Description	Seam No.	Heat No.	Flux Type & Lot No.	Est. Initial RT_{NDT} (°F)	Chemistry		Chem. Factor (F)	Adjustments for 1/4t			
					Cu (wt%)	Ni (wt%)		□)% NDT (°F)	Margin Terms		ART_{NDT} (°F)
									□□ (°F)	□□ (°F)	
L. int Shell Girth Weld	1-344	21935	Linde 1092 #3869	-50	0.183	0.704	172.2	56.7	28.0	0.0	62.7

Based on the information presented in this request and the referenced License Amendment Request with the corresponding NRC Safety Evaluation Report, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 30, 1998 Safety Evaluation Report.

(2) Operator Training and Established Procedures that Limit the Frequency of Cold Over-Pressure Events

The procedures and training used to limit cold over-pressure events are still in use and are materially unchanged from those approved by the NRC in the Safety Evaluation Report for PNPS Relief Request PRR-28 (Reference TAC No. MB6074, Accession No. ML030640204).

5. Duration of Proposed Alternative

The duration of this request is for the extended license period ending June 8, 2032.

6. Precedents

A similar request has been approved for the Peach Bottom Atomic Power Station, Units 1 and 2 (Reference Accession No. ML 112770217).