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

September 15, 2016

Mr. John 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 STATION - ISSUANCE OF RELIEF REQUEST NO. PRR-52 – RELIEF FROM ASME CODE, SECTION XI REQUIREMENTS FOR PRESSURE TESTING OF CLASS 1 PRESSURE RETAINING COMPONENTS AS A RESULT OF REPAIR/REPLACEMENT ACTIVITY AND USE OF CODE CASE N-795 (CAC NO. MF7025)

Dear Sir or Madam:

By letter dated October 21, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15301A761), supplemented by letters dated February 16 and August 2, 2016 (ADAMS Accession Nos. ML16057A177 and ML16224B014 respectively), Entergy Nuclear Operations, Inc. (the licensee) submitted Relief Request No. PRR-52 to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements for pressure testing of Class 1 pressure retaining components as a result of repair/replacement activities at the Pilgrim Nuclear Power Station (Pilgrim).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(2), the licensee requested to use the proposed alternative because compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee proposed to implement provisions that are similar to ASME Code Case N-795 "Alternative Requirements for Boiling Water Reactor (BWR) Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1."

The NRC staff found complying with ASME Code, Section XI, IWB-5221(a) requirement to conduct the pressure testing of Class 1 components identified in this relief request following repair/replacement activities (e.g., in an unscheduled maintenance event or a forced outage event) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. As set forth in the enclosed Safety Evaluation, the NRC staff concludes that the alternatives proposed in Relief Request No. PRR-52 provides reasonable assurance of structural integrity and leak tightness of the Class 1 components. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, pursuant to 10 CFR 50.55a(z)(2), the NRC staff authorizes Relief Request No. PRR-52 at Pilgrim for the Fifth 10-year inservice inspection interval which started on July 1, 2015, and is scheduled to end on June 30, 2025.

If you have any questions, please contact the project manager, Booma Venkataraman, at (301) 415-2934 or 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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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE RELIEF REQUEST NO. PRR-52

ENTERGY NUCLEAR OPERATIONS, INC.

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated October 21, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15301A761), supplemented by letters dated February 16 and August 2, 2016 (ADAMS Accession Nos. ML16057A177 and ML16224B014 respectively), Entergy Nuclear Operations, Inc. (the licensee) submitted Relief Request No. PRR-52 to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements for pressure testing of Class 1 pressure retaining components as a result of repair/replacement activities (e.g., in an unscheduled maintenance or a forced outage event) at the Pilgrim Nuclear Power Station (Pilgrim).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(2), the licensee requested to use the proposed alternative in Relief Request No. PRR-52 because compliance with the specified requirements as described in the request would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee proposed to implement provisions that are similar to ASME Code Case N-795 "Alternative Requirements for Boiling Water Reactor (BWR) Class 1 System Leakage Test Pressure Following Repair/Replacement Activities, Section XI, Division 1."

2.0 REGULATORY REQUIREMENTS

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the Section XI, of editions and addenda of the ASME Code that become effective subsequent to editions specified in paragraphs (g)(2) and(g)(3) of 50.55a and that are incorporated by reference in paragraph (a)(1)(ii) of 50.55a, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(g)(4)(ii), inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must

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comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (a) of 50.55a, 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1", when using Section XI, that are incorporated by reference in paragraphs (a)(3)(ii) of 50.55a), subject to the conditions listed in paragraph (b) of 50.55a. However, a licensee whose inservice inspection interval (ISI) commences during the 12 through 18-month period after July 21, 2011, may delay the update of their Appendix VIII program by up to 18 months after July 21, 2011.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates that (1) the proposed alternatives would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(b)(2)(xxvi), the repair and replacement activity provisions in ASME Code, 1998 Edition of Section XI, IWA-4540(c) for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of 50.55a.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Component Affected

The components affected are ASME Code Class 1 pressure retaining components in containment, excluding the reactor pressure vessel (RPV). The licensee stated that scope of this relief request include:

- Non-isolable welded connections repair/replacement activities
- Non-isolable non-welded mechanical joints repair/replacement activities

3.2 ASME Code Requirements

The ASME Code requirements applicable to this request originate in Section XI, IWA-4000, IWA-5000, and IWB-5000.

- The requirements in IWA-4540(a) states that the repair/replacement activities performed by welding or brazing on a pressure retaining boundary shall include a hydrostatic or system leakage test in accordance with IWA-5000, prior to, or as part of, returning to service;

- In accordance with IWA-5212(a), system leakage tests and system hydrostatic tests shall be conducted at the pressure and temperature specified in IWB-5000, IWC-5000, and IWD-5000.
- In accordance with IWB-5221(a), the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.
- In accordance with IWA-5213(b), a ten minute holding time for non-insulated components, or a four hour holding time for insulated components, is required after attaining test pressure.

The requirements specific to Class 1, 2, and 3 mechanical joints originate in IWA-4540(c) of Section XI to the 1998 Edition of the ASME Code, as mandated by 10 CFR 50.55a(b)(2)(xxvi).

- In accordance with IWA-4540(c) of the 1998 Edition, mechanical joints made in installation of pressure retaining items shall be pressure tested in accordance with IWA-5211(a). Mechanical joints for component connections, piping, tubing (except heat exchanger tubing), valves, and fittings, nominal pipe size (NPS)-1 and smaller, are exempt from the pressure test.

3.3 Applicable Code Edition and Addenda

By letter dated June 19, 2015 (Accession No. ML15166A401), the NRC authorized the licensee to continue to use the 2001 Edition through 2003 Addenda of Section XI for the repair/replacement activities, pressure testing, and nondestructive examination activities until standardized corporate procedures for these activities are updated to the 2007 Edition with 2008 Addenda on December 31, 2017, during the Fifth 10-year ISI interval of Pilgrim (Relief Request PRR-26). The licensee has updated the Pilgrim's code of record for the fifth 10-year ISI interval (started on July 1, 2015, and is scheduled to end on June 30, 2025) to the 2007 Edition with 2008 Addenda of the ASME Code.

3.4 Licensee's Proposed Alternative to the ASME Code

The licensee proposed to perform the pressure test of the Class 1 components identified in this relief request at a reduced test pressure following repair/replacement activities in an unscheduled maintenance event or a forced outage event prior to returning the plant to service. Specifically, the licensee's proposed alternative is to implement provisions that are similar to ASME Code Case N-795. This code case has not been incorporated by reference into 10 CFR 50.55a by inclusion in RG 1.147, Revision 17.

The details of the licensee's proposed alternative pressure test are as follows:

- The licensee proposed to use, at a minimum, the ASME Code Case N-795 specified test pressure (i.e., at least 87 percent of the ASME Code, IWB-5221(a) required pressure to conduct pressure testing.

The licensee stated that the nominal operating pressure at 100 percent rated reactor power at Pilgrim is 1035 pounds per square inch gauge (psig), thus, the proposed test pressure will be at least 900 psig (i.e., at least 87 percent of 1035 psig). The licensee will obtain the proposed test pressure by use of nuclear heat during normal operational start-up sequence.

- The licensee proposed to use a longer hold time than the hold time specified in ASME Code Case N-795. The proposed extended hold time is one hour for non-insulated components and eight hours for insulated components prior to performing the ASME Code required VT-2 visual examinations following pressurization of the components for pressure testing.

The licensee stated that it will not use the proposed reduced test pressure to satisfy the requirements for pressure test of the RPV.

The licensee stated that it will not use the proposed reduced test pressure to satisfy the requirements in Table IWB-2500-1, Category B-P.

3.5 Reason for Request

The licensee stated, in part, in its application that Class 1 pressure tests for repair/replacement activities in accordance with ASME Code, Section XI, IWA-4540 at pressures corresponding to 100 percent rated reactor power when performed after ASME Code, Table IWA-2500-1, Category B-P testing has been completed, requires abnormal plant conditions/alignments. Testing at these abnormal plant conditions /alignments results in additional risks and delays while providing little added benefits. ASME Code Case N-795 is intended to provide alternative test pressure for certain Class 1 pressure tests. This ASME Code Case would be used following repair/replacement activities (excluding those on the reactor vessel) which occur subsequent to the periodic Class 1 pressure test required by ASME Code, Table IWB-2500-1, Category B-P and prior to the next refueling outage on those components that cannot be isolated.

3.6 Licensee's Basis for Use

In the August 2, 2016, letter, the licensee discussed the nuclear safety aspect of pressure test performed at pressure of at least 900 psig as part of a normal start-up evolution. In the normal operational start-up sequence, all systems are in normal alignment and all Core Standby Cooling Systems (CSCS) are available. The licensee stated that the nominal setpoint of the Mechanical Hydraulic Control (MHC) system is set at 90 percent of normal operating pressure. The licensee stated that at 5 percent reactor power, there are two redundant pressure regulators controlling pressure with the turbine bypass valves, bypassing 5 percent steam, within its 25 percent steam capacity. The licensee stated that this is manageable with minimal but steady steam demand above the decay heat contribution. The licensee indicated that this allows the reactor to maintain a balanced condition with water fed from the constant flow Control Rod Drive (CRD) system and rejected within the reactor water cleanup (RWCU) system capacity. This permits safe access to drywell for the component pressure test and the associated VT-2 visual examination.

The license stated that the proposed hold time which is longer than the ASME Code required hold time provides additional time for leakage to be identified during the associated VT-2 visual examination following pressurization at the proposed test pressure of 900 psig. However, the license stated that the ASME Code pressure test at the normal operating pressure of 1035 psig after repair/replacement activities requires abnormal plant conditions and system alignments which create significant operational challenges, and are accompanied with inherent risk that may affect safe operation of the plant. In the August 2, 2016, letter, the licensee tabulated the nuclear safety and risks associated with performance of a pressure test by nuclear heat as compared to performance of pressure test by non-nuclear heat as follows.

Method	Proposed pressure test at 87 percent of normal operating pressure during normal operational startup sequence	ASME Code required pressure test at 100 percent of normal operating pressure (Pilgrim Procedure PNP 2.1.8.5)
Source	Nuclear heat (steam)	Non-nuclear heat (coolant)
RPV and Main Steam Lines essentially water-solid	No	Yes
Heat removal	MHC system	Manual drain to condenser through RWCU
Pressure control	MHC system	Manual control of CRD pump
Level Control	Feed Water Level Control System	Manual balance of drain and CRD with RWCU discharge path
Reactivity Control	Using Control Rods	Control Rods all in
Mode	Startup (Mode 2)	Shutdown (Mode 4)
Normal Interlock	All active	Portions of Primary Containment Isolation System (PCIS) and Reactor Protection system (RPS) Initiation Logic defeated; Recirculation Pump Speed Control interlock defeated
RPV Pressure-Temperature (P-T) Limits	Not affected	Approaching (P-T) limits

The licensee stated that the pressure testing by nuclear heat at proposed pressure of 900 psig during normal reactor start-up evolution with all systems in normal alignment and with a full complement of CSCS available would be a conservative approach when compared to the pressure testing by non-nuclear heat.

3.7 Basis for Hardship

In the August 2, 2016, letter, the licensee stated that existing plant procedures require drywell entry and inspection at low reactor power levels to reduce personnel exposure to radiation. For the ASME Code pressure test during an unscheduled maintenance outage, the licensee estimated that drywell entry with the plant at 100 percent rated power would result in radiation exposure of 919 milli roentgen equivalent man (mrem). However, drywell entry and personnel radiation exposure at the proposed reduced pressure during normal operational startup sequence is estimated to be 25 mrem.

In the August 2, 2016, letter, the licensee discussed hardship associated with risk and safety of plant operation if the Code-compliant pressure test at 100 percent rated power is performed. The licensee's summary is provided below.

Non-nuclear Heat Method

The licensee stated that in an unscheduled short duration maintenance outage, the higher decay heat load from the reactor creates significant challenges to the operations staff if the components listed in this relief request are pressurized to the ASME Code required pressure for pressure testing. For testing, the RPV is filled with coolant and the connecting steam line piping is flooded to the Main Steam Isolation Valves (MSIV) to provide an essentially water-solid condition. This condition removes the availability of both the High Pressure Coolant Injection system and the Reactor Core Isolation Cooling System. The Reactor Recirculation (RRC) system and the CRD system pump along with the reactor decay heat would be used as energy sources to heat-up and pressurize the primary systems to the required normal operating pressure of 1035 psig. The plant only has the RWCU and the manual operation of Main Steam Line (MSL) drains to control coolant level, pressure, and temperature to assist in meeting test conditions, as well as ensuring that test conditions are maintained inside the technical specifications (TS) pressure temperature limits report (PTLR) conditions. Any interruption of RWCU blowdown flow would cause the reactor pressure to increase which may require further operator action to depressurize the RPV.

The licensee also stated that the protective controls such as the RPS, the PCIS, and the Recirculation System speed control interlocks are defeated during testing. The Residual Heat Removal (RHR) Shutdown Cooling (SDC) is intentionally removed from service and isolated since it would exceed its PCIS isolation setpoint as pressure is increased beyond its suction piping limitation of 80 pounds per square inch (psi). The RHR suction piping rating of 80 psi is the limiting system condition, as such, the normal system isolation setpoint is set at a conservative lower value. As a setup for this test, the reactor is scrammed and the mode switch is placed in "SHUTDOWN" position. Isolating RHR shutdown cooling under high decay heat loads requires abnormal plant conditions and alignments, and is accompanied by inherent risk.

The licensee stated that should component malfunction or miss-communication occur during testing, it would be dependent on the operator to use Safety Relieve Valves (SRV) and certain system operation to maintain stable plant condition. Plant procedure specifies the minimum CSCS availability that must be maintained. Based on test setup, the RWCU and MSL drains are the primary depressurization methods which leaves very little margin for error, should this letdown path be lost with a solid vessel to maintain the RPV within the TS PTLR curve.
Nuclear Heat Method

The licensee stated that using the nuclear heat method, Pilgrim cannot use its designed pressure control systems to reach 100 percent normal operating pressure without being near 100 percent reactor power. The reactor at 100 percent rated power creates a drywell environment that will not allow human access for any inspection activities, based on safety concerns from exposure to high radiation and high temperature.

3.8 Duration of the Alternative

The licensee submitted this relief request for the fifth 10-year ISI interval which started on July 1, 2015, and is scheduled to end on June 30, 2025.

3.9 NRC Staff Evaluation of the Alternative

The NRC staff evaluated this relief request pursuant to 10 CFR 50.55a(z)(2). The NRC staff focused on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship.

Hardship or unusual difficulty

The NRC staff agrees that requiring the licensee to comply with IWB-5221(a) to conduct the ASME Code pressure test of the Class 1 components identified in this relief request following repair/replacement activities (e.g., in an unscheduled maintenance event or a forced outage event) would result in significant radiation exposure of personnel involved in performing the Code-compliant pressure test at 100 percent rated reactor power. High radiation dose rates are expected in the drywell with the plant at 100 percent rated reactor power. In addition de-inerting the containment at 100 percent rated power could increase risk to safe operation of the plant. Therefore, complying with the ASME Code required pressure tests would constitute hardship.

The NRC staff also agrees that performing the pressure tests using non-nuclear heat by flooding the main steam lines and the RPV with water to provide an essentially water-solid condition would be undesirable for system operation in a short duration unscheduled maintenance outage. Therefore, the NRC staff determined that the abnormal plant conditions and significant operational challenges and risks from reduced safety margins constitute hardship and unusual difficulty.

Test Pressure

In evaluating the licensee's proposed alternative, the NRC staff assessed whether the licensee used the highest reasonably achievable test pressure to conduct pressure testing of the Class 1 non-isolable welded connections and non-isolable non-welded mechanical joints (excluding the RPV) for leakage following repair/replacement activities in an unscheduled short maintenance outage, and the manner in which the licensee will perform the testing and the associated VT-2 visual examinations. The NRC staff found that:

- The licensee will conduct the pressure test at a pressure of at least 900 psig (which is consistent with 87 percent of pressure required by IWB-5221(a)).
- The test pressure of at least 900 psig will be attained by use of nuclear heat during normal operational start-up sequence.
- The reactor pressure and heat removal are controlled by MHC system which is a normal pressure and temperature control mode for a BWR. The primary emergency core cooling systems are all operable.

The NRC staff determined the proposed alternative pressure testing will accomplish pressure testing without placing the plant in abnormal conditions, without exposing plant personnel to significant radiation, and without challenging the plant's safe operation and personnel safety. Therefore, the NRC staff determines that the licensee's proposed pressure test is adequate because the IWA-5240 required VT-2 visual examination with the proposed extended hold time can reasonably be expected to identify any evidence of leakage following the repair/replacement activities.

Structural Integrity and Leak Tightness

In addition to the analysis described above, the NRC staff considered whether the licensee's proposed alternative provided reasonable assurance of structural integrity and leak tightness of the Class 1 components after repair/replacement activities.

The NRC staff notes that although the proposed test pressure is slightly lower than the ASME Code required test pressure, it still allows for adequate pressurization of the components, and as accompanied with the ASME Code required VT-2 visual examination that will be performed after proposed extended hold time, it allows for detection of potential leakage.

- As part of the pressure test, the licensee will perform the VT-2 visual examination of the components in accordance with the IWA-5240 requirements to identify any through wall leak or mechanical joint leak.
- For non-insulated components, the VT-2 visual examination will be performed after attaining and holding test pressure for one hour to allow for leakage accumulate at the potential leak location and be detected by the examination.
- For insulated components, the VT-2 visual examination will be performed after attaining and holding the test pressure for eight hours to allow for leakage accumulate at the potential leak location and be detected by the examination.

The NRC staff finds that with the above longer hold times, the possibility of observing leakage will be increased, should a through wall leak or a mechanical joint leak occur. The NRC staff also finds that the above hold times exceed the hold times specified in IWA-5213(a) and ASME Code Case N-795 and are, therefore, adequate.

Furthermore, the NRC staff determines that the proposed alternative (i.e., slightly reduced test pressure with increased hold time) provides reasonable assurance that any through wall and/or joint leakage in the Class 1 components after repair/replacement activities will be identified by the VT-2 visual examination, and the licensee will take appropriate action.

Therefore, the NRC staff concludes that the proposed system leakage test performed following the repair/replacement activities on the Class 1 components (excluding the RPV) accompanied by the associated VT-2 visual examinations performed after the proposed extended hold time is adequate to provide reasonable assurance of structural integrity and leak tightness of the components. The NRC staff further concludes that complying with the requirement specified in IWB-5221(a) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff's authorization of the proposed alternative in this relief request neither permits use of this alternative pressure test to satisfy the requirements of Table IWB-2500, Examination Category B-P, nor to satisfy the pressure test requirements of the RPV.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the Class 1 pressure retaining components, and compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the 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)(2). Therefore, the NRC staff authorizes Relief Request No. PRR-52 pursuant to 10 CFR 50.55a(z)(2) at Pilgrim for the Fifth 10-year ISI interval which started on July 1, 2015, and is scheduled to end on June 30, 2025.

The authorization of the proposed alternative in Relief Request PRR-52 does not imply or infer the NRC approval of ASME Code Case N-795.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved by the NRC staff, remain applicable, including third-party review by the Authorized Nuclear In service Inspector.

Principal Contributor: Ali Rezai

Date: September 15, 2016

If you have any questions, please contact the project manager, Booma Venkataraman, at (301) 415-2934 or 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

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