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

February 26, 2016

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (CAC NO. MF7124)

Dear Sir or Madam:

By letter dated November 23, 2015, as supplemented by letter dated January 28, 2016, Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted Relief Request No. 19 (IP2-ISI-RR-19) for the Indian Point Nuclear Generating Unit No. 2 (Indian Point, Unit 2). Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), Entergy requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) related to the system leakage test of Class 1 piping conducted at or near the end of each 10-year inspection interval. Specifically, IP2-ISI-RR-19 proposes an alternative pressure boundary for the ASME Code system leakage test for large bore pipe (> 1 inch), ASME Code Class 1 reactor coolant pressure boundary process, drains, test, flush lines and connections on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject piping segments, and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the staff authorizes the use of the licensee's proposed alternative at Indian Point, Unit 2, for the fourth 10-year inservice inspection interval which is scheduled to end on May 31, 2016.

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. 50-247

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST IP2-ISI-RR-19 REGARDING SYSTEM LEAKAGE TEST OF CLASS 1

PIPING AT OR NEAR THE END OF INSPECTION INTERVAL

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated November 23, 2015 (Agencywide Document Access and Management System (ADAMS) Accession No. ML15342A027), as supplemented by letter dated January 28, 2016 (ADAMS Accession No. ML16036A017), Entergy Nuclear Operations, Inc., the licensee, requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) specifically related to the system leakage test of Class 1 piping conducted at or near the end of each inspection interval. The licensee submitted relief request IP2-ISI-RR-19 for the Indian Point Nuclear Generating Unit No. 2 (Indian Point, Unit 2).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee proposed an alternative pressure boundary for the ASME Code system leakage test on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), the 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 ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 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 during successive 120-month inspection intervals must 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 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 17) when using Section XI, that are incorporated by reference in paragraphs (a)(3)(ii) and (iii) of 50.55a), subject to the conditions listed in paragraph (b) of 50.55a.

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.

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 Components Affected

The components affected are ASME Code Class 1 pipe segments. In accordance with IWB-2500 (Table IWB-2500-1), they are classified as Examination Category B-P, Item Number B15.10.

The licensee identified these pipe segments in Table 1 of the attachment to the relief request, and they are:

- Small size reactor coolant system (RCS) drains, test, and flush lines
- 14 inch diameter piping segment between the residual heat removal (RHR) system inlet motor operated valves (MOV) 730 and 731
- Safety injection system (SIS) loops low head check valves 897A through 897D upstream piping
- SIS loops high head check valves 857A through 857D upstream piping

The materials of construction of the subject pipe segments are austenitic stainless steel. Table 1 of the attachment to the relief request provides details on materials specification for each piping.

3.2 Applicable Code Edition and Addenda

The code of record for the fourth 10-year inservice inspection (ISI) interval is the 2001 Edition through 2003 Addenda of the ASME Code.

3.3 Duration of Relief Request

The licensee submitted this relief request for the fourth 10-year ISI interval which began on March 1, 2007, and is scheduled to end on May 31, 2016.

3.4 ASME Code Requirement

The ASME Code, Section XI, IWB-2500, Table IWB-2500-1, Examination Category B-P, requires the system leakage test to be conducted according to IWB-5220 and the associated VT-2 visual examinations to be performed according to IWA-5240 prior to plant startup following each refueling outage. 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 IWB-5222(a), the pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The required VT-2 visual examination shall, however, extend to and include the second closed valve at the boundary extremity. In accordance with IWB-5222(b), the pressure retaining boundary during system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system pressure boundary.

3.5 Proposed Alternative

The licensee proposed to use the boundary described in IWB-5222(a) with all valves in the position required for normal reactor operation startup when conducting the system leakage test (i.e., pipe segments listed in Table 1 of attachment to the relief request) at or near the end of each inspection interval. Specifically, the proposed alternative is to perform the system leakage test at a reduced pressure (i.e., with all isolation valves in the position required for reactor operation startup). The licensee stated that (a) it will satisfy the pressure holding period of IWA-5213 prior to begin conducting the associated VT-2 visual examination, and (b) it will perform the associated VT-2 visual examination in accordance with IWB-5222(a) and IWA-5240.

3.6 Basis for Alternative

Small size RCS drains, test, and flush lines

The licensee stated that there are nine lines of 2 inch and 3 inch diameter that are used as drains, test, and flush lines. The configurations of these lines consist of isolation valves in series or a valve and a blind flange. In some configurations, the pipe segment may tee to additional lines where an additional valve, cap, or flange provides the second boundary. Pressure testing of these pipe segments at nominal operating pressure would require the opening of the inboard isolation valve at normal operating RCS pressure conditions. Opening the inboard isolation valve is contrary to the design double isolation barrier requirement. The proposed system leakage test will not specifically pressurize the pipe segments past the first isolation valve (i.e., pipes between the isolation valves will not be pressurized) for this inspection. The associated VT-2 visual examination is performed in accordance with IWB-5222(a), IWA-5240, and IWA-5213.

14 inch diameter pipe segment between the RHR system inlet MOV 730 and 731

The licensee stated that this pipe segment consists of 75 feet of 14 inch diameter pipe between the RHR inlet valves 730 and 731. These valves are interlocked at a required setpoint of 365 pound per square inch gauge (psig) to avoid over pressurization of the RHR system. The interlock prevents manual opening of the valves from the Control Room with the RCS pressure above the setpoint. There are no test connection points in this line. The proposed system leakage test will not specifically pressurize the pipe segments past the first isolation valve (i.e., pipes between the isolation valves will not be pressurized) for this inspection. The associated VT-2 visual examination is performed in accordance with IWB-5222(a), IWA-5240, and IWA-5213.

SIS loops low head check valves 897A through 897D upstream piping

The licensee stated that these four pipe segments consist of a 2 inch diameter pipe span between two check valves oriented toward the RCS. These lines are for injecting high head emergency core cooling system (ECCS) water after an accident. The primary and secondary isolation devices are check valves and provide the required double isolation barrier for the reactor coolant pressure boundary (RCPB). Leakage testing of these pipe segments would require plant modifications. The proposed system leakage test will not specifically pressurize the pipe segments past the first isolation valve (i.e., pipes between the isolation valves will not be pressurized) for this inspection. The associated VT-2 visual examination is performed in accordance with IWB-5222(a), IWA-5240, and IWA-5213.

In the January 28, 2016, letter, the licensee provided additional information. This information is as follows.

- The licensee stated that the majority of the subject pipes are insulated. Table 1 of attachment to the January 28, 2016, letter, shows the insulated and uninsulated piping. All pipe segments are accessible for the VT-2 visual examination following system leakage test. The licensee will conduct the VT-2 visual examination in accordance with IWA-5240.

- The licensee stated that the subject pipe segments contain welded connections that are governed by the Indian Point risk informed (RI)-ISI program. Table 1 of the attachment to the January 28, 2016, letter, shows the number and type of welded connections and their susceptibility to possible degradation mechanism.

- The licensee stated that an operating experience (OE) review did not identify any industry or plant specific events where cracking or severe loading is a problem in any of the affected sections of piping located downstream of the normally closed first isolation device. Thermal fatigue of un-isolable piping connected to the RCS is being managed in accordance with the industry guidance in Materials Reliability Program (MRP)-146 "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines" and associated industry interim guidance "EPRI-MRP Interim Guidance for Management of Thermal Fatigue" (ADAMS Accession No. ML15189A100).

- The licensee stated that review of plant-specific OE did not identify any occurrences of inside diameter and outside diameter initiated stress corrosion cracking and through-wall pressure boundary leakage in the subject piping segments.

- The collection and measurement of leakage to the containment are monitored as required by Technical Specification (TS) 3.4.15. The daily plant status report identifies the 24 hour average total RCS leakage and the 24 hour average unidentified leakage based on RCS mass balance calculations each shift. Increasing trends and spikes are observable day to day. Procedures have been established to allow operator response to increasing leak rates and leak rate alarms.

- The licensee stated that pipe segments identified in Items 1 through 9 of Table 1 of the attachment to the January 28, 2016, letter, are lines that are downstream of a normally closed isolation device and are not pressurized. The pipe segments identified in Items 10 through 18 of Table 1 are lines that are relied upon to perform an ECCS function. These lines are pressurized at pressures lower than normal RCS pressure during refueling outage activities or during pump/valve testing. The visual inspections under the Boric Acid Corrosion Control (BACC) Program and walkdowns performed every refueling outage include the pipe segments for which relief is being requested. A VT-2 visual examination is performed during the RCS pressure test with the upstream isolation device in the normally closed position every refueling outage. In addition, the bolted connections in these lines are visually examined every refueling outage with the insulation removed to look for signs of leakage. Therefore, any sign of through-wall leakage in these lines and their associated welded connections would be detected and evaluated in accordance with TSs and appropriate corrective actions would be taken.

3.7 Basis for Hardship

The licensee stated that the Class 1 pipe segments for which relief is requested are equipped with isolation valves which provide double isolation of the RCPB. These valves are generally maintained in the closed position during normal plant operation. The piping outboard of the first isolation valve is not normally pressurized to the RCS pressure. To perform the ASME Code required leakage test, it would be necessary to manually open the inboard valves to pressurize these piping segments. Pressurization by this method defeats the double isolation and reduces the margin of personnel safety for those performing the test. Performing the test with the inboard isolation valves open requires several man-hours to position the valves for the test and restore the valves to their closed positions once the test is completed. These valves are located in close proximity to the RCS loop piping, and thus would require personnel entry into high radiation areas within the containment and a consequent increase in radiation exposure (the estimated dose rate is provided in Table 1 of the attachment to the relief request). Concerns with personnel safety exist due to potential spills associated with use of temporary jumper hoses to bypass the inboard isolation valves. Use of hydraulic test equipment and temporary hoses introduces the possibility of a personnel safety hazard if a connection or hose fails in the presence of inspection personnel. Furthermore, use of hydraulic test equipment or temporary hoses for test locations in the overhead away from normal personnel access areas would require ladders or scaffolding to be installed for the test and inspection. This process would add to the occupational dose associated with pressure testing these lines.

3.8 NRC Staff Evaluation

The NRC staff has evaluated IP2-ISI-RR-19 pursuant to 10 CFR 50.55a(z)(2). The 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, without a compensating increase in the level of quality and safety.

Hardship

The NRC staff found that requiring the licensee to comply with IWB-5222(b) and extend the pressure boundary to all Class 1 components within the system boundary when conducting system leakage tests at or near the end of the ISI interval would result in hardship. The basis for the hardship is as follows. During normal operation, the subject pipe segments listed in Table 1 of the Attachment to the relief request are isolated from the reactor coolant by isolation valves. The valves are designed to serve as a double isolation barrier to the RCPB. The licensee could open the inboard isolation valve manually, bypass the inboard isolation valve by use of a "jumper" around the valve, or use high pressure connections and an external pump to pressurize these piping segments to the RCS operating pressure to perform the required system leakage test. The above actions, however, defeat the double isolation criteria, conflict with the plant design requirements, and reduce safety. The above actions would pose unnecessary safety hazards to plant personnel operating equipment. In addition, a postulated break in any temporary connection could expose personnel to excessive radiation dose since these pipe segments are located in high radiation areas. Furthermore, the licensee would have to redesign the RCS system piping because compliant methods with the plant design do not exist to accommodate performance of the ASME Code leakage tests of the subject pipe segments. As a result, the NRC staff has determined that defeating the double isolation requirements of 10 CFR 50.55a(c)(2)(ii) and modifying existing piping configurations would expose plant personnel to unnecessarily high radiation and safety hazards and would constitute a hardship.

Test Pressure

In evaluating the licensee's proposed alternative, the NRC staff assessed whether the licensee used the highest achievable test pressure to conduct system leakage tests and the manner in which the licensee performed the testing and the associated VT-2 visual examinations. The staff found that the licensee will conduct the leakage tests with all isolation valves in the closed position as required for normal operation and will not pressurize the subject pipe segments. As part of the leakage tests, the licensee will perform the VT-2 visual examinations on the subject pipe segments in accordance with IWA-5240 requirements to identify any leaks or boron residue. The licensee will perform the leakage tests without modifications to existing piping configurations and associated isolation valves. This approach will not: conflict with the plant design requirements; create unnecessary safety hazards; or cause excessive radiation exposure to the personnel involved. Therefore, the staff determined that the licensee's proposed system leakage tests are adequate because it is reasonable to expect the IWA-5240 required VT-2 visual examinations will identify any evidence of leaks or boron residue.

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 subject pipe segments and associated welded connections based on: (1) the presence or absence of known active degradation mechanisms; and (2) the significance of a leak and/or structural failure of the pipe segments and their welded connections.

Presence or absence of known active degradation mechanisms:

The NRC staff notes that the subject pipe segments are made of stainless steel. Potential degradation mechanism of these pipes can include fatigue (both low cycle and high cycle) and stress corrosion cracking (SCC). Low cycle fatigue cracks are known to have relatively slow growth and OE has shown that SCC under the conditions associated with the piping under consideration is not expected. Furthermore, the subject pipe segments and their associated welded connections are governed by the Indian Point RI-ISI program, and the high cycle fatigue (thermal fatigue) cracking of un-isolable piping connected to the RCS is managed by an augmented program. Therefore, it is reasonable to expect that any significant degradation of the piping under consideration would be detected by the alternative leakage test accompanied by the VT-2 visual examinations performed.

Significance of a leak and/or structural failure of the pipe segments and welded connections:

The NRC staff notes that in the unlikely event the subject pipe segments develop a through-wall flaw and leak, the existing Indian Point reactor coolant leakage detection systems would identify the leakage during normal operation, and the licensee will take appropriate corrective actions in accordance with the plant's TSs. In addition, regular walkdowns, the BACC program, and the VT-2 visual examination performed following ASME Code required system leak testing provide additional assurance that any through-wall leaks in the lines would be detected. Therefore, the staff determined that based on the alternative leakage test accompanied by performance of the ASME Code required VT-2 visual examinations, it is reasonable to conclude that if significant service induced degradation occurs, evidence of that degradation will be detected by either the proposed examinations, the RCS leakage detection systems, or other inspections.

Therefore, the NRC staff finds that the proposed system leakage tests accompanied by the VT-2 visual examinations are adequate to provide reasonable assurance of structural integrity and leak tightness of the pipe segments under consideration. Furthermore, the staff finds that complying with the requirements specified in IWB-5222(b) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject pipe segments and complying with the specified ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the staff authorizes the use of the licensee's proposed alternative at Indian Point, Unit 2, for the fourth 10-year ISI interval which is scheduled to end on May 31, 2016.

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

Principal Contributor: Ali Rezai, NRR

Date: February 26, 2016

requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the staff authorizes the use of the licensee's proposed alternative at Indian Point, Unit 2, for the fourth 10-year inservice inspection interval which is scheduled to end on May 31, 2016.

Sincerely,

/RA/

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

Docket No. 50-247

Enclosure:
Safety Evaluation

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