February 8, 2006

Mr. Michael R. Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - RELIEF REQUEST (RR) NO. 74 (TAC NO. MC7307)

Dear Mr. Kansler:

By letter dated June 8, 2005, as supplemented by letters dated October 27, 2005, and December 5, 2005, Entergy Nuclear Operations, Inc. (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 1989 Edition, for the system hydrostatic test requirements for the Indian Point Nuclear Generating Unit No. 2. The relief request proposed a system leakage test to the normal operating pressure boundary rather than a hydrostatic test to the full ASME Code Class 1 pressure boundary.

The Nuclear Regulatory Commission staff has concluded that the proposed alternatives to the ASME Code requirements in RR No. 74 are acceptable, and that compliance with the specified ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The results are provided in the enclosed safety evaluation. Pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternatives are authorized for the remainder of the third 10-year inservice inspection interval, which currently ends on December 31, 2006.

If you have any questions regarding this approval, please contact the Indian Point Project Manager, John Boska, at 301-415-2901.

Sincerely,

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosure:
As stated

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 74

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NUMBER 50-247

1.0 INTRODUCTION

By letter dated June 8, 2005, Agencywide Documents Access and Management System (ADAMS) accession number ML051660264, Entergy Nuclear Operations, Inc. (the licensee) submitted a relief request to the Nuclear Regulatory Commission (NRC) for Indian Point Nuclear Generating Unit No. 2 (IP2). The submittal requested relief from selected requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, Table IWB-2500-1, Examination Category B-P, which requires a system hydrostatic test to include all ASME Code Class 1 components. The licensee provided additional information in its letters dated October 27, 2005, and December 5, 2005, ADAMS accession numbers ML053080244 and ML053490199.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) 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(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service 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. The regulations require that inservice examination 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 incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Code of record for IP2 is the 1989 Edition of Section XI of the ASME Code, with no addenda. In response to an NRC request for additional information, the licensee confirmed that the IP2 third 10-year ISI interval started on July 1, 1994, and will end on December 31, 2006. The licensee also stated that this interval has been extended due to outages greater than 6 months and to coincide with a refueling outage as allowed by the 1989 Edition of the ASME Code paragraphs IWA-2430(e) and IWA-2430(d), respectively.

3.0 TECHNICAL EVALUATION

The information provided by the licensee in support of the request for relief from ASME Code requirements has been evaluated and the basis for disposition is documented below.

3.1 ASME Code Requirements

Examination Category B-P, Item B15.50, requires that a system hydrostatic test be performed on Class 1 components at or near the end of each ISI interval. The pressure retaining boundary during the test shall include all Class 1 components within the system boundary. The test pressure, as required by Paragraph IWB-5222(a), is required to be between 102% and 110% of the nominal operating pressure associated with 100% rated reactor power and corresponding to the system temperature during the test, as specified in Table IWB-5222-1.

3.2 Licensee's ASME Code Relief Request

In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the pressure test requirements for portions of piping in the safety injection (SI) and residual heat removal (RHR) systems that connect to the reactor coolant system (RCS) (see Table 3.2 below for descriptions of the piping segments included in this alternative). The licensee's alternative is to perform the hydrostatic tests at pressures less than those specified by the ASME Code based on the hardship that would be incurred if the ASME Code-required pressures are imposed.

Table 3.2 - Piping Segments in Request for Relief RR-74

Segment Description	Code Category	Schedule/Diameter	Length
Regenerative Heat Exchanger Flush taps.	B-P	Sch 160/3" Dia	< 1ft
Regenerative Heat Exchanger Flush taps.	B-P	Sch 160/3" Dia	< 1ft
Regenerative Heat Exchanger Flush taps.	B-P	Sch 160/3" Dia	< 1ft
Regenerative Heat Exchanger Flush taps.	B-P	Sch 160/3" Dia	< 1ft
Regenerative Heat Exchanger Flush taps.	B-P	Sch 160/3" Dia	< 1ft
Reactor Coolant System Loop Drain Lines	B-P	Sch 160/2" Dia	1ft
Reactor Coolant System Loop Drain Lines	B-P	Sch 160/2" Dia	1ft
Reactor Coolant System Loop Drain Lines	B-P	Sch 160/2" Dia	1ft
Reactor Coolant System Loop Drain Lines	B-P	Sch 160/2" Dia	1ft

Table 3.2 - Piping Segments in Request for Relief RR-74

Segment Description	Code Category	Schedule/Diameter	Length
Residual Heat Removal Line from the Reactor Coolant System	B-P	Sch 140/14" Dia	75 ft
Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System	B-P	Sch 140/10" Dia Sch 160/6" Dia Sch 160/2" Dia	28 ft 2 ft 1 ft
Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System	B-P	Sch 140/10" Dia Sch 160/6" Dia	12 ft < 1 ft
Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System	B-P	Sch 140/10" Dia Sch 160/6" Dia Sch 160/2" Dia	10 ft 12 ft 3 ft
Safety Injection and Residual Heat Removal Lines to the Reactor Coolant System	B-P	Sch 140/10" Dia Sch 160/6" Dia	18 ft < 1 ft
Safety Injection Lines to the Reactor Coolant System	B-P	Sch 160/2" Dia	87 ft
Safety Injection Lines to the Reactor Coolant System	B-P	Sch 160/2" Dia	61 ft
Safety Injection Lines to the Reactor Coolant System	B-P	Sch 160/2" Dia	37 ft
Safety Injection Lines to the Reactor Coolant System	B-P	Sch 160/2" Dia	15 ft

3.2.1 Licensee Basis for Relief

The piping segments listed in Table 3.2 are connected directly to the reactor coolant system, and, in accordance with the reactor coolant pressure boundary definition in 10 CFR 50.2, are classified as ASME Code Class 1 up to and including the second isolation valve. Each of these piping segments, except for the RHR system piping, is isolated from the RCS by a self-actuating check valve designed to prevent reactor coolant from escaping the RCS, while providing a passive injection flow-path for coolant injection. The use of check valves in these piping segments for isolation from the RCS prevents, by design, their pressurization by the primary RCS, and conversely, their pressurization to any pressure greater than that in the RCS.

The RHR piping segment is also connected directly to the RCS; however, this piping is isolated from the RCS by two in-series motor-operated valves (MOVs). These MOVs are interlocked to ensure redundant isolation of the RCS from the lower design pressure (600 pounds per square inch gage [psig]) RHR system. Plant operating instructions require that these MOVs be closed when the RCS pressure exceeds 350 psig.

During performance of the Section XI inservice hydrostatic pressure test, the RCS would be brought to system normal operating pressure of approximately 2235 psig, at which time the subject piping segments are isolated from the RCS by their respective check valves, or other valves in the RHR segment. No method currently exists for pressurizing these piping segments to full test pressure during the Section XI hydrostatic pressure test.

Two methods that the licensee investigated are: (1) the use of temporary high pressure hoses connected to RCS test connections, vent or drain piping to “jumper” around the isolation check valves, and (2) the use of hydrostatic pumps connected to each piping segment. Both of these methods conflict with plant design requirements and 10 CFR 50.55a(c)(ii) by eliminating the double isolation boundary required for the reactor coolant pressure boundary when the reactor vessel contains nuclear fuel. The use of either of these methods would require a redesign of the RCS and the installation of new piping designed to meet the plant construction code and licensing commitments. This option is cost prohibitive and imposes a burden to the licensee which is not commensurate with the increase to plant safety achieved through compliance with the ASME Code, Section XI pressure test requirement versus use of the proposed alternative test method.

The purpose of the ASME Code, Section XI pressure test is to detect existing through-wall defects in the pressure-retaining boundary by the identification of leakage from the boundary. The detection of pressure boundary leakage from such through-wall defects can be achieved at pressures lower than the pressure associated with 100% rated reactor power.

3.2.2 Licensee’s Proposed Alternative Examination

The proposed alternate testing method will achieve the highest test pressure in each piping segment listed in Table 3.2 that can be achieved without plant modification, and while continuing to comply with plant Technical Specifications (TSs) and design requirements when nuclear fuel is contained in the reactor. The Section XI test procedure requires a holding time (4 hours for insulated components and 10 minutes for non-insulated components) after attaining test pressure in order to allow sufficient fluid leakage to collect to ensure detection by the visual, VT-2, examination. The alternate testing method would reduce the amount of leakage from a through-wall defect, however, it would not be expected to prevent detection of a leak during a visual, VT-2, examination.

The piping segments from the high pressure and intermediate pressure SI and the SI accumulators will be pressurized using the SI pumps to approximately 1450 psig which is the pressure achieved with the SI pumps running in the minimum recirculation flow mode.

The piping segments from the RHR system segment will be pressurized to approximately 350 psig and visually examined when the RHR system is providing shutdown cooling during plant startup following the refueling outage.

Based on the hardships associated with costly plant modifications and redesign, IP2 considers the proposed alternative test method to be acceptable for satisfying pressure boundary integrity of the segments identified in Table 3.2 while maintaining compliance with plant design requirements, plant TSs and the requirement of 10 CFR 50.55a(a)(c)(ii). Sufficient test pressure in conjunction with the test pressure holding time will allow detection of any leakage from the pressure-retaining boundary of the subject piping segments. Accordingly, the licensee requests relief from the ASME Code in accordance with 10 CFR 50.55a(a)(3)(ii).

3.3 Evaluation

The ASME Code requires that a system hydrostatic test be performed once each interval to include all Class 1 components within the RCS boundary. The hydrostatic test must be performed at or near the end of the ISI interval, and the test pressure is required to be between 102% and 110% of the nominal operating RCS system pressure associated with 100% rated reactor power, depending on the system temperature during the test. However, several piping line segments are connected to the RCS through self-actuating check valves or interlocked MOVs, which does not allow normal RCS pressure to be used to pressurize these segments. In order to test the subject piping segments to normal operating RCS pressure (approximately 2235 psig), the licensee would have to make plant design modifications to enable the use of high pressure hoses as temporary jumpers around valves or employ hydrostatic pumps connected directly to the piping segments. Either of these options would conflict with plant TSs and operational design requirements by potentially defeating the RCS boundary double isolation, which is mandated when fuel is present in the reactor vessel. To require the licensee to make plant modifications in order to pressurize the subject line segments to normal RCS pressure would result in a considerable hardship.

Pressure testing of the RCS is typically performed during the return to power sequence at the end of a refueling outage using reactor coolant pumps and pressurizer heaters to bring the RCS to normal operating temperature and pressure, prior to initiating core criticality. At this time, the subject SI and RHR piping segments are isolated from the RCS. These segments are described in Table 3.2, and primarily consist of limited runs of piping between the first and second isolation valves in the SI connections on each of the four primary coolant loops. In addition, a section of RHR piping between the first and second isolation valves is also included. The piping segments are fabricated of austenitic stainless steel and range in diameter from 2 to 14-inch nominal pipe size (NPS) (see Table 3.2 for specific sizes and wall thicknesses). These segments, including the first and second isolation valves, are considered part of the reactor coolant pressure boundary, as defined in 10 CFR 50.2.

For SI piping segments connecting to RCS Loops 1 through 4, the self-actuating isolation check valves are designed to prevent back-flow of primary coolant into the respective high and low pressure SI piping, while providing a passive flow-path for injecting coolant during normal start-ups and shutdowns, as well as during postulated emergency events. Therefore, the design and function of these valves do not allow piping upstream of the first isolation check valve in each line segment to experience normal RCS pressures. In order to subject the identified piping segments to RCS pressure, the first isolation valve would have to be bypassed. This would require the licensee to make pressure boundary modifications to the existing piping to accommodate fittings, valves, or other appurtenances needed to support this activity. Another option would be for the licensee to use a stand-alone hydrostatic pump connected to the subject piping between the first and second isolation valves to obtain a pressure equivalent to that during normal RCS operation. Again, this may require modifications to the piping pressure boundary, and could potentially inject water into the primary system if pump pressure slightly exceeds normal RCS pressure. Either of these methods would result in a significant hardship for the licensee.

Similar problems exist for the RHR piping segment, which has redundant isolation from the RCS by two interlocked MOVs. The RHR system has a maximum design pressure of 600 psig and is normally only operated during shutdown and start-up sequences. The MOVs are closed and locked prior to the RCS pressure exceeding 350 psig, therefore the RHR piping segment cannot be pressurized during a normal RCS pressure test sequence.

As an alternative to pressurizing the subject line segments in accordance with the ASME Code requirements noted above, the licensee has proposed the following:

- For the subject SI piping line segments, use the safety injection pumps running at minimum recirculation mode, to pressurize segments to approximately 1450 psig.
- For the subject RHR line segment, visually examine the piping when RHR is operating at 350 psig during plant start-up following the refueling outage.

The licensee’s proposal represents the highest test pressures that can be obtained without significant plant modifications and are intended to test the subject piping segments to conditions similar to those that may be experienced during postulated design-basis events. The NRC staff agrees that the proposed test pressures will be sufficient to produce detectable leakage from significant service-induced degradation sources, should these exist, as well as verify that connections in these piping segments that may have been opened during the outage have been properly secured. The licensee has also committed to meeting the hold times for insulated (4 hours) and non-insulated (10 minutes) components, as shown in paragraph IWA-5213, prior to performing the required VT-2 visual examinations.

The NRC staff determined that the ASME Code requirements would be a significant hardship for the licensee to perform. The licensee would have to make plant design modifications to enable the use of high pressure hoses as temporary jumpers around these valves or employ hydrostatic pumps connected directly to the piping segments. Either of these options would conflict with plant TSs and operational design requirements by potentially defeating the RCS boundary double isolation.

It is concluded that to require the licensee to pressurize the subject piping segments in accordance with the ASME Code requirements noted above would require significant plant modifications and would subject the licensee to a hardship or unusual difficulty without a compensating increase in the level of quality. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee’s proposed alternative is authorized.

4.0 CONCLUSION

The NRC staff has reviewed the licensee's submittal and concludes, for Request for Relief RR-74, that compliance with the ASME Code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. The alternative proposed by the licensee provides reasonable assurance of the continued leak integrity or structural integrity of the subject components. Therefore, Request for Relief RR-74 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval at IP2. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested and approved remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: N. Ray

Date: February 8, 2006