September 5, 2007

Mr. Michael A. Balduzzi
Sr. Vice President & COO
Regional Operations, NE
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - RELIEF REQUEST NO. RR-01 (TAC NO. MD4695)

Dear Mr. Balduzzi:

By letter dated February 28, 2007, Entergy Nuclear Operations, Inc. (the licensee), submitted a relief request for the fourth 10-Year Inservice Inspection Interval. The request is to visually examine the upper circumferential and longitudinal pressurizer welds during each refueling outage for evidence of leakage during system pressure tests in lieu of volumetric examinations. This request is essentially identical to the previous third 10-Year Inservice Inspection Interval relief request RR-07.

Inservice inspection of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (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 Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). Also, 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.

Based on the enclosed safety evaluation, the NRC staff concludes that the proposed relief request is acceptable and approves the request to visually examine the upper circumferential and longitudinal pressurizer welds during each refueling outage for evidence of leakage during system pressure tests in lieu of volumetric examinations.

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

Sincerely,

/RA/

Mark G. Kowal, 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: See next page

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

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF NO. RR-01

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated February 28, 2007, Entergy Nuclear Operations, Inc. (the licensee), submitted Relief Request RR-01 for the fourth 10-Year Inservice Inspection (ISI) Interval for Indian Point Nuclear Generating Unit No. 2 (IP2). The request is to visually examine the upper circumferential and longitudinal pressurizer welds during each refueling outage for evidence of leakage during system pressure tests in lieu of volumetric examinations. This request is essentially identical to the previous third 10-Year ISI Interval relief request RR-07.

2.0 REGULATORY REQUIREMENTS

ISI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (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 Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). Also, 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 ASME Code of record for the IP2 fourth 10-year interval

Enclosure

inservice inspection program, which began on March 1, 2007, is the 2001 Edition with the 2003 Addenda of Section XI of the ASME Code.

3.0 EVALUATION

RR-01

ASME Code Component Identification

Pressurizer Shell-to-Head Circumferential and Longitudinal Welds PZRC-5 and PZRL-4

ASME Code Requirement

ASME Code, Section XI, Table IWB-2500-1, Category B-B, Item Numbers: B2.11 (circumferential weld) and B2.12 (longitudinal weld) require essentially 100% volumetric examination coverage of the weld length per ASME Code, Section XI, Figures IWB-2500-1, IWB-2500-2, and IWB-2500-20(b).

ASME Code Case N-460, *Alternative Examination Coverage for Class 1 and Class 2 Welds*, as an alternative approved for use by the NRC in Regulatory Guide (RG) 1.147, Revision 14, *Inservice Inspection Code Case Acceptability*, Section XI, Division 1 states that a reduction in examination coverage due to part geometry or interference for any ASME Class 1 or 2 weld is acceptable provided that the reduction is less than 10%, i.e., greater than 90% examination coverage is obtained.

ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item 15.10 requires a system leakage test to be conducted prior to plant start up following a reactor refueling outage. Code Case N-498-4 *Alternative Requirements for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems,* Section XI, Division 1 proposes as an alternative to the ASME Code requirements that a system leakage test may be conducted at or near the end of inspection interval prior to reactor start up.

Licensee's Basis for Relief Request (As stated)

> Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that compliance with the [ASME] Code requirement is impractical. The pressurizer (PZR) was designed and fabricated to [industry design codes and Federal regulations] in effect during the late 1960's and did not incorporate the clearances needed for the examination of these welds. [When the pressurizer was installed, other items were installed in the following sequence: insulation; concrete biological shield; and the piping at the top of the pressurizer was welded in the annular area between the vessel and the biological shield. The distance between the pressurizer vessel and the biological shield is about eight inches, of which half is filled with asbestos insulation]. The [industry design codes and Federal regulations] used at the time did not provide for full access for inservice inspection as required by later [Editions of the ASME Code, and Federal regulations.]
>
> The upper circumferential (PZRC-5) and longitudinal (PZRL-4) welds are enclosed in a biological and missile shield and are therefore completely inaccessible for volumetric examination. The shell-to-bottom head circumferential and longitudinal welds (PZRC-1

and PZRL-1) will be examined in accordance with [ASME Code, Section XI,] IWB-2500-1 requirements.

In order to gain access to the PZR shell-to-head welds to perform the [ASME] Code required volumetric examinations, the PZR system and piping would have to be re-designed (which includes re-routing of the piping at the top of the PZR, removal of the asbestos insulation on the PZR, and the removal and enlargement of the concrete biological shield surrounding the PZR). This type of system modification can only be accomplished with an extended shutting down of the plant.

It is expected that any through-wall defects would be detected by this examination prior to the failure of the pressurizer based on the expectation that the component will experience leakage before a catastrophic failure (“leak before break").

The level of inspections proposed for the Fourth [10-year ISI] interval has been in effect for the first three [10-year ISI] intervals. Based on the reliable operating history of this IP2 vessel, and similar vessels at other [pressurized-water reactor] PWR plants, the performance of VT-2 examinations for leakage, granting of this relief request will not decrease the overall level of quality and safety of this component.

Licensee’s Proposed Alternative Examination (As stated)

Entergy proposes to visually examine (VT-2) the upper circumferential (PZRC-5) and longitudinal (PZRL-4) welds each refueling outage for evidence of leakage during system pressure tests performed in accordance with [ASME Code, Section XI, Table] IWB-2500-1, Category B-P, and [ASME] Code Case N-498-4.

4.0 NRC STAFF EVALUATION

The ASME Code requires that essentially 100% volumetric examination coverage be achieved for the pressurizer shell-to-head welds PZRC-5 and PZRL-4. The licensee is unable to volumetrically examine essentially 100% of the subject weld lengths and has requested relief from the ASME Code volumetric examination requirements. As an alternative to the ASME Code, the licensee has proposed to perform VT-2 visual examinations for leakage on the subject welds each refueling outage in accordance with the ASME Code and ASME Code Case N-498-4.

The IP2 PZR was designed and fabricated to design requirements in effect during the late 1960's which did not require the same consideration of accessability of the subject welds for inservice examination that is found in later ASME Code editions and Federal regulations. The licensee noted that when the PZR was installed and insulated, the concrete biological shield was installed and all piping was then welded in the annular area between the vessel and the biological shield wall. There is approximately 8 inches of space between these components; however, half of this space is filled with asbestos insulation and is a hazard to the licensee’s personnel. However, the licensee noted that it will be able to volumetrically examine PZR shell-to-bottom head circumferential weld PZRC-1 and longitudinal weld PZRL-1. If significant service-induced degradation were occurring, there is reasonable assurance that evidence of it would be detected by these examinations. If degradation of welds PZRC-1 and PZRL-1 is not detected, this will provide an indication of the integrity of welds PZRC-5 and PZRL-4.

PZR upper circumferential weld PZRC-5 and longitudinal weld PZRL-4 are enclosed in a biological and missile shield which makes the subject welds inaccessible for volumetric examination. The NRC staff determined that in order for the licensee to gain access to the subject welds to perform the ASME Code-required volumetric examinations, the PZR and piping would have to be redesigned, including the removal of the asbestos insulation and the removal and modification of the concrete biological shield surrounding the PZR. These modifications would place a burden on the licensee. Therefore, the NRC staff determined that these ASME Code requirements are impractical.

General Design Criteria 30 of Appendix A to 10 CFR Part 50 requires means for detecting and to the extent practical, identifying the location of the source of reactor coolant system leakage. According to the IP2 Improved Technical Specifications (TS) Bases, Section B3.4.15, Revision 2 dated August 15, 2006, the IP2 leakage detection system has the capability of detecting leakage increases of 0.5 to 1.0 gallons per minute (gpm). The licensee’s capability of detecting leakage increases of 0.5 to 1.0 gpm is similar to standard industry practice. In addition, the IP2 TS 3.4.15 requires that the licensee monitor atmospheric radioactivity in the containment, which would provide an indication of reactor system coolant leakage. In either case, the licensee will be able to detect a leak in time to take corrective action according to TSs and plant operating procedures.

Therefore, the licensee’s proposed alternative to perform VT-2 visual examinations of the subject welds each refueling outage in accordance with the ASME Code and ASME Code Case N-498-4, the plant’s leakage and radiation monitoring systems, and volumetric examinations of welds PZRC-1 and PZRL-1 provide reasonable assurance of the leak tightness of the PZR welds.

5.0 CONCLUSION

The NRC staff has reviewed the licensee's submittal and concludes that ASME Code requirements are impractical for the subject welds listed in relief request RR-01, Revision 0, for the IP2 fourth 10-year ISI interval. Furthermore, the NRC staff concludes that the licensee’s proposed VT-2 visual examinations, the plant’s leakage and radiation monitoring system, and volumetric examinations of the PZR shell-to-bottom head circumferential weld PZRC-1 and longitudinal weld PZRL-1 provide reasonable assurance of the leak tightness of the PZR welds. Therefore, relief request RR-01, Revision 0, is granted, pursuant to 10 CFR 50.55a(g)(6)(i), for the IP2 fourth 10-year ISI interval.

The NRC staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) for relief request RR-01, Revision 0, is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Thomas K. McLellan

Date: September 5, 2007