Entergy

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July 19, 2007

Re: Indian Point Unit 2
Docket No. 50-247
NL-07-082

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: **Supplemental Submittal Regarding Relief Request RR-02, "Proposed Alternative for Regenerative Heat Exchanger Welds," and RR-05, "Risk-Informed ISI (Relief from B-F & B-J examination requirements)." (TAC Nos. MD4696 and MD4700)**

REFERENCES: 1. Entergy Letter dated February 28, 2007, P.W. Conroy to Document Control Desk, "4th Ten-Year Interval Inservice Inspection and Containment Inservice Inspection Program Plan at Indian Point Unit 2 (IP2)"

Dear Sir or Madam:

By letter dated February 28, 2007 (Reference 1) Entergy Nuclear Operations, Inc. submitted the 4th Ten-Year Interval Inservice Inspection and Containment Inservice Inspection Program Plan for the period March 1, 2007 through April 3, 2016 for IP2. Appendix B of the enclosure contained seven (7) relief requests. The NRC staff requested additional information via Teleconference on June 19, 2007 in order to complete its review of Relief Requests RR-02 and RR-05. The purpose of this letter is to provide the responses to the questions discussed at the Teleconference. Responses to NRC questions on RR-02 and RR-05 are provided in Attachment 1 to this letter.

If you have any questions or require additional information, please contact Mr. T.R. Jones, Manager, Licensing at (914) 734-6670.

Sincerely,

T. R. Jones
Licensing Manager
Indian Point Energy Center

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NRR

Attachments:

1. Response for Additional Information Regarding Relief Request RR-02 and RR-05 (TAC Nos. MD4696, MD4700)

cc: Mr. John P. Boska, Senior Project Manager, NRC NRR DORL
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 2
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-07-082

**Response for Additional Information Regarding
Relief Requests RR-02 and RR-05
(TAC Nos. MD4696 and MD4700)**

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

Response for Additional Information Regarding
Relief Request RR-02, "Proposed Alternative for Regenerative Heat Exchanger Welds."
(TAC No. MD4696)

Responses to the questions discussed at June 19, 2007, Teleconference regarding Relief Request RR-02, "Proposed Alternative for Regenerative Heat Exchanger Welds." (TAC No. MD4696) are as follows:

1. **Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), Request RR-02 proposes an alternative to the examination requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Subarticle IWB-2500 on the basis that the alternative examination will provide an acceptable level of quality and safety. Request RR-02 specifically proposes that the provisions of ASME Code Case (CC) N-706, "Alternative Examination Requirements of Table IWB-2500-1 and Table IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchangers, Section XI, Division 1," be used as an alternative to the examination requirements of Table IWB-2500-1, Examination Categories B-B and B-D for the specified regenerative heat exchanger (RHX) 21 vessel welds.**

   CC N-706 states that the alternative examination requirements specified in Table 1 of the CC may not be applied to any heat exchanger nor to any heat exchanger design or configuration that has experienced a through wall leak. Note (1) from Table 1 of the CC states that the application of the alternative examination requirements of Table 1 of the CC is limited to those welds that are part of the as-received RHX assembly. The RHX assembly may be formed from multiple smaller heat exchanger subcomponents connected by sections of piping. According to Note (1), all of the smaller heat exchanger subcomponents and the connecting piping are considered to be within the boundary of the RHX assembly.

   Section E of RR-02 states that a leak occurred in January 2004 at San Onofre Nuclear Generating Station (SONGS), Unit 3 in the letdown line exiting the RHX. Please indicate whether the design or configuration of the RHX at SONGS 3 corresponds to that of RHX 21 at Indian Point 2. If the design or configuration of the RHX at SONGS 3 does correspond to the design or configuration of RHX 21 at Indian Point 2, please provide additional detail regarding how the leak that occurred in the SONGS 3 RHX letdown line was determined to be outside the scope of CC N-706, taking into consideration the provisions of Note (1) under Table 1 of the CC and the definition of RHX 21 at Indian Point 2 as an RHX assembly which includes all of the smaller heat exchanger subcomponents and the connecting piping.

Response

The event at SONGS Unit 3 identified a small leak that developed in a pipe weld on the letdown outlet side of the Regenerative Heat Exchanger. Vibration resulting from operation of Positive Displacement Charging Pumps with inadequate discharge pressure pulsation dampeners, and a locked spring can hanger was the primary cause. Initial inspections and system monitoring revealed that the charging line into the RHX and the letdown line out of the heat exchanger were vibrating at elevated levels. The leak was from the weld on the Letdown piping outlet at the bottom of the vertically oriented RHX.

The SONGS Unit 3 Regen Heat Exchanger is of different design, manufacturer, and configuration then at Indian Point Unit 2. The SONGS Unit 3 RHX is of Whitlock design with a vertical orientation.

The IP2 RHX is designed and manufactured by Atlas Industrial Mfg Corp., for Westinghouse, and is of horizontal configuration. In addition, IP2 installed suction stabilizers and discharge pulsation dampers on each of the charging pumps to reduce vibration-induced fatigue on the pump and associated piping.

2. CC N-706, Table 1, Note (2) states that all welds under Item No. 1.10 of the table shall have received at least one volumetric examination and that the preservice or construction code volumetric examination may be used to meet this requirement. The interpretation of this requirement is that, for a particular component, all items that would be subject to the alternative examination requirements of Table 1 must have received this volumetric examination.

Section E of RR-02 states that the pressure retaining welds in RHX 21 had received at least one volumetric preservice examination. Please verify that all welds in RX 21 that are within the scope of the proposed alternative had previously received this volumetric examination.

For all welds identified in RR-02 that have received a volumetric examination, please state whether these examinations included only the preservice examinations required by Subarticle IWB-2200 or additional successive inspections during previous Inservice Inspection (ISI) intervals required by Subarticle IWB-2400. Please discuss any relevant conditions that were found during these examinations.

Response

A pre-service volumetric Inspection was performed on all Regenerative Heat Exchanger Head-to-shell, Tube sheet-to-shell and nozzle-to-vessel welds no recordable indications noted. Additional successive examinations were performed on these welds during the 2^{nd} and 3^{rd} interval which ended in 2000 in accordance with IWB-2400 and approved Relief Requests. Eleven (11) were volumetrically examined while the remainders were surfaced examined with no recordable indications identified during the inspections.

3. **Please discuss the programs that are currently in place for Reactor Coolant System (RCS) leakage monitoring in the vicinity of RHX 21.**

Response

Unidentified leakage is monitored in accordance with IP2 Improved Technical Specifications. Additionally, the containment atmosphere particulate radioactivity is monitored per Technical Specification requirements.

The heat exchanger will also continue to receive a system leakage test (Procedure 2-PT-R75) prior to startup after each refueling outage. During this system test, the components receive a visual (VT-2) examination. The corresponding piping and component supports will also continue to be inspected per the requirements of the Code, as they are not affected by this relief request.

Response for Additional Information Regarding
<u>Relief Request RR-05, "Risk-Informed ISI (Relief from B-F & B-J exam. requirements)."</u>
(TAC No. MD4700)

Responses to the questions discussed at June 19, 2007, Teleconference regarding Relief Request RR-05, "Risk-Informed ISI (Relief from B-F & B-J exam. requirements)." (TAC No. MD4700) are as follows:

1. **The newer versions of the American Society of Mechanical Engineers (ASME) Code have reduced the exempted portions of high pressure safety injection piping from nominal pipe size (NPS) 4 to NPS 1.5 for volumetric examination. This reduction in exempted piping has caused other licensees to add ASME Class 1 pressurized water reactor (PWR) high pressure safety injection piping to the scope of their risk-informed inservice inspection (RI-ISI) programs, and to implement their chosen RI-ISI methodology to classify, risk-rank, and to select, as necessary, additional locations for the next ISI interval. Please describe how you treated this issue in your RI-ISI program for the fourth 10-year ISI interval when you updated your code of record from the 1989 edition to the 2001 edition with 2003 addenda.**

   <u>Response</u>

   For the fourth 10-year ISI interval Indian Point Unit 2 included the high pressure safety injection piping with nominal pipe size $\geq$1.5 for volumetric examination in the scope of the Risk-Informed In-service inspection (RI-ISI) program. This is described in the original submittal for the Third ISI Interval (SER TAC No. MC0624).

2. **The submittal states that the number and locations of inspections have not changed and therefore the original changes in risk estimate remain valid. The change in risk estimate may change because of changes in the probabilistic risk assessment (PRA) and/or degradation mechanism even if the risk ranking does not change enough to require changing of the inspection program. Please explain why the previous changes in risk estimate remain valid considering changes that have been made to the PRA and degradation mechanism.**

   <u>Response</u>

   The inputs from the updated probabilistic risk assessment (PRA) model were reviewed against inputs used in the original RI-ISI consequence assessment. The PRA inputs reviewed include initiating event conditional core damage probability and system unavailabilities (e.g. valve failure rates). The impact on the consequence ranking results is shown below.

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2006 Update Information

Initiator	IE Freq	CDF	CCDP	Description	RI-ISI Consequence Input (1) CCDP	Rank	Impact
IE-T3	1.87E+00	1.53E-06	8.20E-07	TURBINE TRIP WITH MAIN FEEDWATER AVAILABLE	1.22E-06	Medium	None
IE-T2	3.86E-01	8.70E-07	2.26E-06	LOSS OF MAIN FEEDWATER			N/A
IE-A	5.00E-06	1.58E-08	3.15E-03	LARGE LOCA	6.07E-03	High	None
IE-S1	4.00E-05	1.16E-07	2.91E-03	INTERMEDIATE LOCA	3.02E-03	High	None
IE-S2	5.00E-04	3.90E-07	7.80E-04	SMALL LOCA	2.81E-04	High	None
IE-T4	4.27E-04	7.08E-09	1.66E-05	MAIN STEAM LINE BREAK INSIDE CONTAINMENT			N/A

Failure Rates

Hourly Rate		Yearly Rate		Description		Impact
5.36E-07	Per Hr	4.70E-03	Per Yr	CHECK VALVE FAILS TO REMAIN CLOSED	2.12E-02	None
3.00E-06	Per Hr	2.63E-02	Per Yr	AOV FAILS TO REMAIN CLOSED	1.86E-03	None (3)
3.00E-09	Per Hr	2.63E-05	Per Yr	MANUAL VALVE FAILS TO REMAIN CLOSED	8.12E-04	None
		3.00E-03	Per Demand	MOV FAILS TO CHANGE POSITION	1.59E-03	None (3)
		1.00E-03	Per Demand	CHECK VALVE FAILS TO CLOSE (2)	3.77E-05	None (3)
		1.00E-03	Per Demand	AOV DOES NOT CLOSE	8.20E-04	None (3)

(1) = Scientech Report No. 17184-02, Consequence Analysis of Class 1 Piping in Support of Code Case N-578, Indian Point 2 Nuclear Power Plant

(2) = Representative of several system specific values

(3) Per the above, several valve failure rates from the updated PRA are higher with than that used in the RI-ISI application. Per Reference 8, these values were used in segments determined to be Medium consequence rank (See C-1). As such, the following bounds any impact of these changes and shows that the RI-ISI consequence rank does not change:

Upper Bound LOCA CCDP = 3.15E-03
Upper Bound Valve Failure = 3.00E-03

CCDP * Valve = 9.45E-06 or a Medium consequence rank

The results of this review concluded, there are no required changes to the RI-ISI consequence ranking due to the updated PRA input. That is, high consequence segments remained high and medium consequence segments remained medium.