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November 6, 2002
IPN-02-088

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, DC 20555-0001

SUBJECT: Indian Point Nuclear Generating Unit No.3
Docket No. 50-286
Reply to Request for Additional Information
Regarding Proposed License Amendment for
1.4% Measurement Uncertainty Recapture Power Uprate

REFERENCE: 1. Entergy letter to NRC, IPN-02-041, "Request for License Amendment for 1.4% Measurement Uncertainty Recapture Power Uprate," dated May 30, 2002.

Dear Sir:

This letter provides additional information requested by the NRC regarding the license amendment request submitted by Entergy Nuclear Operations, Inc (ENO) in Reference 1. The additional information was requested by the NRC during conference calls with ENO personnel on October 28 and 30, 2002.

The requested information is provided in Attachment I, including a corrected page for the analysis report submitted as Attachment III in Reference 1. The information provided herein does not alter the conclusions of the no significant hazards evaluation previously provided in Reference 1. There are no new commitments identified in this letter. If you have any questions or require additional information, please contact Mr. Kevin Kingsley at 914-734-5581.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 11-6-02

Very truly yours,

JRKelley for

Robert J. Barrett
Vice President, Operations – IP3
Indian Point 3 Nuclear Power Plant

cc: next page
Attachment: as stated

A001

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ATTACHMENT I TO IPN-02-088

RESPONSE TO NRC QUESTIONS REGARDING

PROPOSED LICENSE AMENDMENT REQUEST FOR

1.4% MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

The following questions were provided by NRC during conference calls with ENO personnel on October 28 and 30, 2002.

Question 1:

Section 7.7.2 of the evaluation report (Attachment III to licensee submittal IPN-02-041, dated May 30, 2002) describes the impact of the 1.4% power uprate on steam generator tube structural integrity based on the existing plant condition of 0% tube plugging. Describe the method that will be used to ensure that the evaluation will be reassessed in the event that additional tubes are plugged in the future.

Response 1:

Updating the Indian Point 3 Final Safety Analysis Report (FSAR) is a required implementing action, following NRC approval of the license amendment request for power uprate. Changes to the FSAR will include adding a discussion of the steam generator structural integrity evaluation and the assumption regarding tube plugging. Existing administrative procedures are in place to ensure that changes to the facility as described in the FSAR are evaluated with respect to 10CFR50.59. Therefore existing administrative processes and addition of relevant information to the FSAR will ensure that future tube plugging is evaluated including the potential affect on the steam generator structural integrity evaluation.

Question 2:

Section 7.2 evaluates the impact of the 1.4% power uprate on neutron irradiation of the reactor vessel. The results provided in Table 7-6 regarding upper shelf energy do not appear to be consistent with the fluence values provided by the licensee.

Response 2:

Entergy has confirmed that an error was made in recording the 'projected upper shelf energy (USE) decrease' values from Regulatory Guide 1.99. The corresponding projected end-of-life (EOL) USE values are therefore also incorrect. A corrected page for Table 7-6 is provided at the end of this attachment. The limiting value for projected EOL USE is for the lower shell plate B2803-3 and is based on surveillance capsule data. The corrected Table also includes this clarification. These corrections do not alter the conclusion in the original submittal that the requirements of 10CFR50 Appendix G are met for the proposed power uprate.

Question 3:

Section 7.7.3 discusses the impact of the 1.4% power uprate on flow-induced vibration and tube wear in the steam generators. However, the submittal does not include a discussion of fluidelastic effects.

Response 3:

The summary of results for the tube wear evaluation provided in Section 7.7.3 included the effects of turbulent flow mechanisms, as a result of increased feedwater flowrates, on steam generator tube wear and vibration. The evaluation of fluidelastic flow conditions and vortex shedding concluded that the proposed power uprate would not result in a significant increase in the projected level of tube wear.

The maximum fluidelastic stability ratio was evaluated for the straight leg and U-bend regions. The result for the limiting location, which is the U-bend region, was an increase in the current ratio of 0.7 to 0.71 for the proposed power uprate. This result remains well below the allowable limit of 1.0.

The maximum vibration-induced displacement due to turbulence excitation was evaluated for power uprate. An increase from 6.7 mils to approximately 7.0 mils was determined for the most limiting tube support condition. Displacements of this magnitude will not result in tube-to-tube contact. The projected increase in tube wear reported in Section 7.7.3 (from current 1.3 mils to approximately 2 mils for power uprate) includes the effects of turbulent flow mechanisms. The projected increase in wear continues to remain below the tube wear allowance of 3 mils. The steam generator inspection program required by the plant technical specifications provides a means for monitoring actual tube wear.

Question 4:

Section 7.3 evaluates the impact of the 1.4% power uprate on the reactor internals. Table 7-7 reports the results of the fatigue evaluation and references 'NG-3222.2'. Please explain why this code section is referenced and which version of the code is used for this Table.

Response 4:

The reactor vessel internals are not part of the reactor coolant system pressure boundary and are not governed by the ASME Boiler and Pressure Vessel Code. However, the code does provide a suitable reference for stress analysis acceptance criteria. Previous evaluations of these components used methods and material allowables based on the code. The current analysis references subsection NG (and associated Appendices) from the 1986 edition (with no addendum) of ASME Section III.

TABLE 7-6 PREDICTED EOL (27.1 EFPY) USE CALCULATIONS FOR ALL THE BELTLINE REGION MATERIALS					
Material	**Weight % of Cu**	**1/4T EOL Fluence (1019 n/cm2)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease(a) (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell Plate B2802-1	0.20	0.535	102	25	77
Intermediate Shell Plate B2802-2	0.22	0.535	97	26	72
Intermediate Shell Plate B2802-3	0.20	0.535	95	25	71
Lower Shell Plate B2803-1	0.19	0.535	72	24	55
Lower Shell Plate B2803-2	0.22	0.535	94	26	70
Lower Shell Plate B2803-3	0.24	0.535	68	18(b)	55
Intermediate and Lower Shell Weld Longitudinal Weld Seams (Heat 34B009)	0.19	0.535	112	28	80
Intermediate to Lower Shell Circumferential weld Seams (Heat 13253)	0.22	0.535	111	31	77

Notes:

a. Values are deduced from Figure 2 of NRC Regulatory Guide 1.99, Revision 2, Predicted Decrease in Upper Shelf Energy as a function of Copper and fluence.

b. Calculated using Surveillance Capsule Data from Indian Point Unit 3 Surveillance Capsules T, Y and Z