Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
295 Broadway, Suite 1
PO Box 249
Buchanan, NY 10511-0249
Tel 914 734 5340
Fax 914 734 5718

Fred Dacimo
Vice President, Operations

May 2, 2003
NL-03-076

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

SUBJECT: Indian Point Nuclear Generating Unit No.2
Docket No. 50-247
Reply to Request for Additional Information
Regarding Proposed License Amendment for
1.4% Measurement Uncertainty Recapture Power Uprate (TAC MB6950)

REFERENCES:

1. Entergy letter to NRC, NL-02-155, "Proposed Changes to Technical Specifications: Measurement Uncertainty Recapture Power Uprate, Increase of Licensed Thermal Power (1.4%)", dated December 12, 2002.

2. Entergy letter to NRC, NL-03-058, "Reply to Request for Additional Information Regarding Proposed License Amendment for 1.4% Measurement Uncertainty Recapture Power Uprate", dated April 3, 2003.

Dear Sir:

This letter provides additional information requested by the NRC regarding the proposed license amendment 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 April 10 and 16, 2003.

The requested information, provided in Attachment I, supplements the material previously submitted in References 1 and 2, and does not alter the conclusions of the no significant hazards evaluation.

A001

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 5-2-03.

Very truly yours,

for F. Dacimo

Fred R. Dacimo
Vice President, Operations
Indian Point Energy Center

cc:

Mr. Patrick D. Milano, Senior Project Manager
Project Directorate I,
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop O 8 C2
Washington, DC 20555

Mr. Hubert J. Miller
Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
Indian Point Unit 2
U.S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

Mr. Peter R. Smith, Acting President
New York State Energy, Research
and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Dept. of Public Service
3 Empire Plaza
Albany, NY 12223

ATTACHMENT I TO NL-03-076

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. 2
DOCKET NO. 50-247

Question 1:

In 1997, NRC issued Amendment 188 to the Indian Point 2 Facility Operating License, regarding use of the Westinghouse generic Best Estimate (BE) large-break (LB) loss-of-coolant-accident (LOCA) analysis evaluation model, EM MOD 7A, Revision 1. The NRC Safety Evaluation (March 31, 1997) for this license amendment identified three conditions of the NRC approval to apply that version of the model for IP2 analyses. Please verify that these conditions remain satisfied for the proposed 1.4% MUR uprate.

Response 1:

The three conditions of the referenced NRC Safety Evaluation remain satisfied as described below.

Condition 3.0.a states: "The version of the EM may be referenced only for the initial IP2 analyses for as long as they remain applicable per 10 CFR 50.46 requirements or until they are superseded by updated analyses. Future analyses using the EM must be performed entirely using the W BE LB LOCA EM MOD 7A Rev.1 version or other fully approved LB LOCA EM."

This condition remains satisfied because the existing LB LOCA analysis, initially performed with EM MOD 7A Rev 1, remains applicable and bounding for the proposed 1.4% power uprate. An updated analysis for the uprate conditions was not required as stated in Table 8-1 of the Power Uprate Application Report (Attachment III to ENO letter NL-02-155).

Condition 3.0.b states: "The imprecision of the correction must be tracked in IP2 10 CFR 50.46 reports as a permanent change or error."

ENO reviewed past annual 50.46 reports beginning in 1998, which first identified application of the NRC approved BE LB LOCA methodology to IP2 analyses. ENO verified that the reported licensing basis Analysis-of-Record PCT (2152 ^{o}F) includes the required correction value attributed to the version of the evaluation model used for the existing IP2 LB LOCA analyses.

Condition 3.0.c states: "Reference to the June 13, 1996 letter must be maintained in appropriate licensing documentation (e.g., technical specifications and / or COLR)."

Although the subject letter is not referenced in the Technical Specifications or the COLR, the requirements of the letter are maintained in appropriate licensing documentation. The analysis of record for the IP2 BE LB LOCA is documented in WCAP 13837 Rev 1, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for Indian Point 2 Nuclear Plant," dated December 1996. This WCAP, referenced in the IP2 FSAR, incorporates the re-analysis work plan described in the June 13, 1996 letter. Also, this letter is referenced in the input documents that support the 10 CFR 50.46 annual reports.

Question 2:

Verify that controls are in place to assure that LOCA analyses bound plant operating conditions.

Response 2:

ENO also addressed this request to support NRC review for use of the BE LB LOCA methodology to Indian Point 3 (reference ENO letter to NRC, NL-03-044 dated March 12, 2003). The following statement provided in that submittal is also applicable to IP2; "ENO and Westinghouse have ongoing processes which assure that the range and values of LOCA analyses inputs for Peak Clad Temperature sensitive parameters bound the as-operated plant ranges and values for those parameters."

Question 3:

The ENO response previously provided to Question 23 of the NRC Request for Additional Information dated March 11, 2003 stated that balance-of-plant piping systems were acceptable for uprate thermal, pressure, and flow rate change factors of less than or equal to 1.05. Please explain the bases for using 1.05 as an acceptance value.

Response 3:

The 1.05 value was used as a screening criterion for comparison to the thermal, pressure, and flowrate change factors that were calculated to quantify the effect of the proposed power uprate on piping system operating conditions. Use of 1.05 is based on the availability of margin in pipe stress and pipe support calculations due to the conservatism inherent in the analytical methods applied in the existing calculations for pre-uprate conditions. A sample of representative calculations was reviewed as part of the uprate feasibility evaluation for IP2. The sample consisted of 4 main steam line calculations (MS-01 through MS-04) and 4 main feedwater calculations (FW-05 through FW-08). The review verified that the analysis methods used are simplified linear elastic computer techniques that are typical of generally accepted industry practice for evaluating piping system acceptability. More sophisticated analytical methods, such as detailed finite-element models, are available which would demonstrate that the margin to design limits is greater (by at least 5 percent) than that demonstrated using the existing linear elastic techniques. The 1.05 value was used as a screening tool to determine if additional more detailed evaluations of existing piping system calculations might be needed as a result of the power uprate operating conditions. The thermal, pressure, and flow rate change factors for uprate conditions were previously reported in the response to Question 23. Change factors were typically 1.00 to 1.01 with a maximum value of 1.02 in two cases involving extraction steam. Therefore, using the 1.05 screening criterion, a determination was made that the existing piping system designs were acceptable for uprate conditions, without performing additional detailed analyses.