March 11, 2003

Mr. Michael R. Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING LICENSE AMENDMENT REQUEST FOR POWER UPRATE (TAC NO. MB6950)

Dear Mr. Kansler:

By application dated December 12, 2002, the Entergy Nuclear Operations Inc., submitted a license amendment request that would revise the operating license and technical specifications for the Indian Point Nuclear Generating Unit No. 2, to allow the use of a more accurate flow measurement instrumentation to allow the licensed core thermal power to be increased by 1.4 percent from 3071.4 megawatts thermal to 3114.4 megawatts thermal.

The Nuclear Regulatory Commission (NRC) staff has reviewed your December 12, 2002, application and concluded that it does not provide technical information in sufficient detail to enable the staff to make an independent assessment regarding the acceptability of the proposal in terms of regulatory requirements and the protection of public health and safety. Enclosed is the NRC staff's request for additional information (RAI).

We have discussed this with your staff and it was agreeable to your staff to respond to this RAI and provide comments within 30 days from receipt of this letter. If you have questions regarding this letter or are unable to meet this response schedule, please contact me by phone on (301) 415-1457 or by electronic mail at PDM@nrc.gov.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager, Section 1
Project Directorate 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

1.4 PERCENT MEASUREMENT UNCERTAINTY RECAPTURE

ENTERGY NUCLEAR OPERATIONS, INC. (ENO)

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1. Please provide a listing of the Indian Point Unit 2 (IP2) instrumentation uncertainty components used as inputs to the reactor power uncertainty calculation, including their associated measurement uncertainties and uncertainty values with respect to power. Discuss the methodology used and show that the mathematical combination of these uncertainties is less than the stated 0.6 percent (with Caldon Leading Edge Flow Meter (LEFM) Check Flow Elements installed) for IP2.

2. The Nuclear Steam Supply System Operating Point parameters for power uprate conditions were calculated for a core power uprate of 1.4 percent (3,114.4 MWt). Provide a listing of the Final Safety Analysis Report Chapter 14 transients and accidents analyses which incorporate these uprate operating point parameters. For those that do not, provide justification that the current values used in the analyses are bounding.

3. Please provide a quantitative discussion confirming that the Low Temperature Overpressure Protection Relief valves have adequate relief capacity to remove the additional decay heat generated by the 1.4 percent power uprate such that there is no increase in peak pressure for this transient. Include a discussion of the NRC-approved methodology used to perform this analysis.

4. With respect to the impacts of the proposed power uprate on the nuclear, thermal-hydraulic and fuel rod design analyses, please provide a listing of the NRC-approved codes and methodologies used for the design analyses discussed in Section 7.10 of the Attachment III of the submittal and confirm that all parameters and assumptions to be used for analyses described in Sections 7.10 of the Attachment III remain within any code limitations or restrictions.

5. Provide a more detailed anticipated transient without scram (ATWS) evaluation that is applicable to IP2 at power uprate conditions to demonstrate that the peak primary system pressure will not exceed the ASME Stress Level C limits of 3200 psig. Justify that the assumptions for the analyses are adequate as they relate to input parameters such as the initial power level, current fuel enrichment, moderator temperature coefficient (MTC), pressurizer safety and relief valves capacity, reactor coolant system volume, steam generator pressure, auxiliary feedwater (AFW) flow rate and its actuation delay time, and the setpoint for the

ATWS Mitigation System Actuation Circuitry (AMSAC) system to actuate the AFW and trip the turbine. The submittal should include a discussion and applicable values of the unfavorable exposure time for the MTC assumed in the analyses. Explain why the Technical Specification value of MTC less than zero would assure the assumed MTC value in the ATWS analysis.

6. Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and revision 1, NSAL 02-4 and NSAL 02-5, to document the problems with the Westinghouse designed steam generator (SG) water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15, 2002, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG measurements and affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation and affect the high-high level trip setpoint. NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how IP2 accounted for these uncertainties documented in these advisory letters in determining the SG water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the loss-of-coolant accident (LOCA) and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still limiting.

7. Upon reviewing LBLOCA models for power uprates, the NRC has recently found plants that require changes to their operating procedures because of inadequate hot leg switch-over times and boron precipitation modeling. Demonstrate that your LBLOCA model continues to comply with 10 CFR 50.46 during the switch-over from the refueling water storage tank to the Containment Sump. Also, discuss how your analyses account for boric acid buildup during long-term core cooling; and discuss how your predicted time to initiate hot leg injection corresponds to the times in your operating procedures.

8. For LOCA and non-LOCA transients and accidents that already assume 2 percent uncertainty in the current safety analysis, please provide discussion on the effects of the change of initial plant conditions for the power uprate to the results of these analyses.

9. Section 8.3.4.2 of the report indicated that loss of flow and locked rotor events were evaluated with respect to departure from nucleate boiling ratio (DNBR). Your evaluation concluded that the existing statepoints for these events remain valid with the exception of the norminal core heat flux, which increases due to the power uprate. Therefore, the higher nominal core heat flux must be applied

to the power statepoints. The analyses with the revised statepoints showed that the DNB design basis remains satisfied. Please provide more details of these evaluation/analysis including the calculated minimum DNBR for these events.

10. Section 8.3.6.5 of the report indicates that the Excessive Load Increase event was evaluated to demonstrate that the DNB design limit is met. Please provide details of this evaluation.

11. Provide a quantified evaluation of the impacts of the 1.4 percent power uprate on the ability of IP2 to cope with a Station Blackout event.

12. Describe the method used for determining the proposed maximum allowable power range neutron flux high setpoints for various number of inoperable main steam safety valves.

13. In the first paragraph of Section 3.5, ENO stated that the LEFM Check System was originally installed in 1980 and the upgrade to the electronic unit, which meets the requirements of the approved Topical Report ER-80P, was installed in October 2002. The second paragraph of this section states that the Caldon LEFM Check System was installed in the fall of 2002. Please explain how the LEFM hardware (spool piece, etc) installation requirements of ER-80P was met in 1980 while ER-80P was approved in 1999. Also, please identify and explain if there was any failure of the LEFM system or its component since its original installation at IP2.

14. In Section 3.6, ENO stated that uncertainty calculations have been performed and determined a mass flow accuracy of better than 0.5 percent of rated flow for IP2. Please submit this calculation for staff review. Additionally, the instrument uncertainty of feedwater flow used in WCAP-15904-P is much lower (proprietary) than the calculated value determined to be better than 0.5 percent of rated flow. It is noted that the instrument uncertainty of feedwater flow used in power calorimetric uncertainty calculation for IP3 (WCAP-15824) was much higher (proprietary) than the calculated value (proprietary) provided in the ENO letter to the NRC, dated November 20, 2002. It is not clear why IP3 power calorimetric calculations used much higher than the calculated value of the LEFM measurement uncertainty while a similar calculation for IP2 used much lower than the calculated 0.5 percent, which makes it non-conservative. Please explain.

15. Section 3.3 provides justification for continued operation of IP2 at the power level of the proposed uprate power with an LEFM Check System out of service. In this section, it is stated that IP2 is operated based on alternate plant instrument, which is benchmarked to the LEFM's last good reading as soon as the LEFM Check System becomes unavailable. This alternate instrumentation has been subject to programmatic, extensive trending relative to LEFM flow and temperature outputs. This section also states that while the accuracy of the alternate instruments may degrade over time, it is considered likely that any degradation as a result of nozzle fouling, drift, and the like, would be imperceptible for the 7-day period as long as steady state conditions persist.

Extrapolating from the programmatic trending data, or otherwise, please quantify the effects of the nozzle fouling and drift in terms of the percent uprate power during the proposed 7-day allowed outage time of an inoperable LEFM.

16. In Section 3.7, ENO stated that loops 21 and 22 LEFM Check Systems were calibrated at Alden Research Laboratory while loops 23 and 24 calibration coefficients are based upon ARL testing of a population of 7 flow elements with similar inside diameters and dimensions. It is assumed that the ARL calibration of loops 21 and 22 LEFM was performed on the plant-specific piping configuration. Please confirm. Staff review of the ARL report of loops 21 and 22 LEFM calibration and loops 23 and 24 LEFM measurement uncertainty calculations, similar to the one submitted in your letter to the NRC dated November 20, 2002, for IP3, is needed to complete our evaluation of the proposed power uprate of IP2.

17. In Section 3.4, ENO stated that all other instrument components that provide fluid condition data for calculation of rated thermal power is controlled, calibrated, and performance monitored to the conditions represented in the overall calorimetric uncertainty evaluation done for the IP2 1.4 percent power uprate. Please confirm IP2 plant procedures for these actions that address all five items of section 1.1.F in RIS 2002-03.

18. Provide in detail the effect of the power uprate on the environmental qualification of electrical equipment.

19. Provide details about the grid stability analysis including assumptions and results and conclusions for the power uprated condition.

20. In Section 7.4.1, Fatigue evaluation has not been performed for RCL piping except the pressurizer surge line because then Code B31.1-1955 did not require such evaluation. Please explain why the evaluation is not applicable to the power uprated conditions given the fact the later ASME Section III Code requires such evaluation.

21. In reference to Section 7.4.2 of Attachment 3 to the amendment request, you stated that the 1.4 percent power uprate does not significantly affect any of the loads applied to the reactor coolant loop piping, steam generator and reactor coolant pump supports resulting from the Snubber Reduction Program. Therefore, you concluded that the design basis of the supports as reconciled for the IP2 Snubber Reduction Program remains applicable for the 1.4 percent power uprate. Provide a technical basis or quantitative evaluation for your conclusion. Also, confirm that the existing design basis analysis support loads has sufficient safety margin to accommodate the load increase due to the proposed 1.4 percent power uprate at IP2.

22. In reference to Table 7-6, “IP2 1.4% Power Uprate Evaluation Summary – Primary-and-Secondary-Side Components,” you indicated that for tube and tube-sheet weld, the reference analysis used conservative high fatigue strength reduction factor with elastic stresses in the fatigue evaluation since primary

stresses exceed 3Sm. Provide a summary describing the reference analysis and the high fatigue strength reduction factor that was used in the analysis. Also, provide the existing design basis stresses and the calculated stress at the uprated condition, that are not shown in the table, for the tube to tube-sheet weld, the divider plate and the tube-sheet/shell junction.

23. In reference to Section 10.9, “Balance-of-Plant (BOP) Piping and Support Evaluation,” you indicated that the changes in operating parameters such as temperature, pressure, and flow rate were determined to be insignificant and you concluded that they have a negligible effect on the existing piping system qualifications. No specific pipe stress re-analysis were required to document the acceptability of the 1.4 percent power uprate conditions. Provide a summary of your quantitative evaluation to demonstrate that there exist sufficient safety margins to accommodate the changes due to the proposed power uprate on the Balance-of-Plant piping and supports.

24. In reference to Section 12.2.5, “Safety-Related Motor Operated Valves,” you evaluated the effect of the proposed power uprate on the motor-operated valves (MOVs) program at Indian Point 2 (IP2) for Generic Letter (GL) 89-10 and GL 95-07 regarding pressure locking and thermal binding or safety-related power-operated gate valves. Provide a summary evaluation of the effects of the proposed power uprate on your response to GL 96-06 regarding overpressurization of isolated piping segment.