August 26, 2002

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE, INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. MB5297)

Dear Mr. Kansler:

In a letter dated May 30, 2002, Entergy Nuclear Operations, Inc. (ENO) submitted a proposed amendment to the Facility Operating License and Technical Specifications (TSs) for Indian Point Nuclear Generating Unit No. 3. Specifically, the proposed amendment would revise the license and TSs to increase the licensed core thermal power level to 3067.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3025 MWt.

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the information provided in the May 30 submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). During a telephone call with the NRC on August 22, 2002, the ENO staff indicated that a response to the RAI would be provided within 30 days.

If you should have any questions, please do not hesitate to call me.

Sincerely,

/RA/

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

Docket No. 50-286

Enclosure: RAI

cc w/encl: See next page

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cc:

Mr. Jerry Yelverton
Chief Executive Officer
Entergy Operations
1340 Echelon Parkway
Jackson, MS 39213

Mr. Robert J. Barrett
Vice President - Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P. O. Box 308
Buchanan, NY 10511-0308

Mr. Dan Pace
Vice President Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. James Knubel
Vice President Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Joseph DeRoy
General Manager Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P. O. Box 308
Buchanan, NY 10511-0308

Mr. John Kelly
Director - Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Charlene Fiason
Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Harry P. Salmon, Jr.
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. James Comiotes
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P.O. Box 308
Buchanan, NY 10511-0308

Mr. John Donnelly
Licensing Manager
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
P.O. Box 308
Buchanan, NY 10511-0308

Mr. John McCann
Manager, Licensing and Regulatory Affairs
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 2
295 Broadway, Suite 1
P. O. Box 249
Buchanan, NY 10511-0249

Resident Inspector’s Office
U.S. Nuclear Regulatory Commission
295 Broadway, Suite 3
P.O. Box 337
Buchanan, NY 10511-0337

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

Mr. John M. Fulton
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

cc:

Ms. Stacey Lousteau
Treasury Department
Entergy Services, Inc.
639 Loyola Avenue
Mail Stop: L-ENT-15E
New Orleans, LA 70113

Mr. William M. Flynn, President
New York State Energy, Research, and
Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. J. Spath, Program Director
New York State Energy, Research, and
Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. Paul Eddy
Electric Division
New York State Department
of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Ray Albanese
Executive Chair
Four County Nuclear Safety Committee
Westchester County Fire Training Center
4 Dana Road
Valhalla, NY 10592

Mr. Ronald Schwartz
SRC Consultant
64 Walnut Drive
Spring Lake Heights, NJ 07762

Mr. Ronald J. Toole
SRC Consultant
Toole Insight
605 West Horner Street
Ebensburg, PA 15931

Mr. Charles W. Hehl
SRC Consultant
Charles Hehl, Inc.
1486 Matthew Lane
Pottstown, PA 19465

Mr. Alex Matthiessen
Executive Director
Riverkeeper, Inc.
25 Wing & Wing
Garrison, NY 10524

Mr. Paul Leventhal
The Nuclear Control Institute
1000 Connecticut Avenue NW
Suite 410
Washington, DC, 20036

Mr. Karl Copeland
Pace Environmental Litigation Clinic
78 No. Broadway
White Plains, NY 10603

Jim Riccio
Greenpeace
702 H Street, NW
Suite 300
Washington, DC 20001

Request for Additional Information

Regarding Proposed Amendment for 1.4% Power Uprate

Measurement Uncertainty Recapture

Indian Point Nuclear Generating Unit No. 3 (IP3)

In a letter dated May 30, 2002, Entergy Nuclear Operations, Inc. (ENO or the licensee) submitted a proposed amendment to revise the license and Technical Specifications (TSs) for IP3 to increase the rated thermal power by recapture of measurement uncertainty by utilizing the Caldon leading edge flow meter (LEFM) check system. The licensee addressed the items in NRC Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" on Feedwater Flow Measurement Technique and Power Measurement Uncertainty in Attachment 3 of the May 30, 2002 letter. However, the NRC staff has determined that additional information is needed as indicated in the following questions:

1. List the 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. Provide data showing that the mathematical combination of these uncertainties is less than the stated 0.6% (with Caldon LEFM check flow elements installed) for IP3.

2. In Attachment 3, Section 3.3, paragraph 8, the licensee states "If the plant experiences a power change of greater than 10% during the seven-day period, the permitted maximum power level would be reduced upon return to full power, in accordance with the power levels described below, since a plant transient may result in calibration changes of the alternate instrumentation."

    a. Describe the significance of the 10% power change limit

    b. Confirm that the stated 10% power changes do not involve power increases above the steady state power level at the time of Caldon LEFM check system becomes inoperable.

3. In Attachment 3, Section 3.3, paragraph 4, the licensee states, "While recognizing that 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 seven-day period as long as steady-state conditions persist." What is the measurement uncertainty, in terms of thermal power, of the alternate instruments, over the seven-day period?

4. In Attachment 3, Section 3.4, paragraphs 6, 7, and 9, the licensee discusses the control of software and hardware configuration of the Caldon LEFM Check system but does not mention other instrumentation that affect the power calorimetric. Provide a discussion of the control of software and hardware configuration of other plant instrumentation that affect the power calorimetric (See NRC RIS 2002-03 Item I.1.F).

Enclosure

5. Attachment 3, Section 3.7, the licensee states, "The Caldon LEFM Check flow element calibrations were based upon Alden Research Laboratory (ARL) testing of a population of seven flow elements with identical inside diameters and dimensions." Were the flow elements tested in an IP3 plant-specific configuration at ARL? Provide the reference and results of the tests.

6. Attachment 3, Section 4.1 references WCAP-15824, "Power Calorimetric for the 1.4% Uprating for Entergy Nuclear Indian Point Unit 3." Submit the referenced WCAP-15824 to NRC for staff review.

7. The nuclear steam supply system operating point parameters for the power uprate conditions were calculated for a core power of 3,067.4 MWt. List the Final Safety Analysis Report (FSAR) 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.

8. 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% 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.

9. With respect to the impacts of the proposed power uprate on the nuclear, thermal-hydraulic and fuel rod design analyses, list the NRC-approved codes and methodologies used for the design analyses discussed in Section 7.10. Confirm that all parameters and assumptions to be used for analyses described in Section 7.10 remain within any code limitations or restrictions.

10. Provide a more detailed anticipated transient without scram (ATWS) evaluation that is applicable to IP3 at power uprate conditions to demonstrate that the peak primary system pressure will not exceed the American Society of Mechanical Engineers 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 circuit 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 TS value of MTC less than zero would assure the assumed MTC value in the ATWS analysis.

11. 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, affecting 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, affecting 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. These 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 IP3 accounts for these uncertainties as documented in the NSALs 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.

12. While reviewing large-break loss-of-coolant accident (LBLOCA) models for power uprates, the NRC staff 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 IP3 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 the analyses account for boric acid buildup during long-term core cooling. Discuss how the predicted time to initiate hot leg injection corresponds to the times in the IP3 operating procedures.

13. Section 6.1.3 indicated that the post-LOCA containment sump temperature performance has been determined to be unaffected by the proposed power uprate. Provide a discussion to support the above conclusion.

14. For LOCA and non-LOCA transients and accidents that already assume 2% uncertainty in the current safety analysis, discuss the effects of the change of initial plant conditions for the power uprate on the results of these analyses.

15. Section 8.3.4.2 indicated that for loss of flow and reactor coolant pump shaft seizure events were evaluated with respect to departure from nucleate boiling ratio (DNBR). The analysis showed that the DNBR design basis remains satisfied. Provide additional details of these evaluation/analysis including the calculated minimum DNBR for these events.

16. Section 8.3.6.5 of the report indicates that the excessive load increase event was evaluated to demonstrate that the DNBR design limit is met. Provide details of this evaluation.

17. Provide a justification for the proposed changes in Table 3.3.2.1, Note C (from 110% full steam flow to 120% full steam flow).

18. Provide the results of an evaluation of the impacts of the 1.4 percent power uprate on the ability of IP3 to cope with a Station Blackout event.

19. In Section 7.2.1 the licensee stated: "The calculated fluences used in this 1.4% power uprate evaluation comply with NRC Regulatory Guide [RG] 1.190". Explain how and why the calculated fluence satisfies the guidance in RG 1.190.

20. In Section 7.7.5, “Regulatory Guide 1.121 Analysis,” the licensee summarized the results of an analysis it performed to define the structural limits for various regions of the SG tube. The section refers to RG 1.121 which provides guidance on calculating the allowable tube repair limit (i.e., utilizing the structural limit, a growth allowance and eddy current measurement uncertainty allowance). However, the licensee did not conclude whether the revised structural limits affect the allowable tube repair limit currently documented in the TSs. Identify the appropriate tube repair limit, and discuss the basis for reaching this conclusion. If the tube repair limit currently documented in the TSs needs to be modified, submit an appropriate TS change.

21. In Section 7.7.3, “Tube Wear,” the licensee described the potential effects of the 1.4% power uprate on SG tube wear. Discuss the potential effects of the 1.4% power uprate on other modes of SG tube degradation (e.g., axial and/or circumferential cracking, pitting, etc.). Factor into your discussion the impact from the increase in primary system hot-leg coolant temperature.

22. Discuss the impact the power uprate will have on the required frequency of SG tube inspections.

23. Section 7.7.2, “Structural Integrity Evaluation,” describes the impact of the power uprate on SG tube structural integrity. The licensee stated that the 1.4% power uprate structural evaluation was performed for 3082 MWt NSSS power and 0% SG tube plugging. Table 2.1 describes three different sets of design parameters, one of which relates to the 0% plugging assumption. Summarize the results of the structural evaluation performed for the other two sets of design parameters, or explain the basis for performing the SG tube structural integrity evaluation for only one set of design parameters.

24. In reference to Section 7.7.2 of Attachment 3 to the amendment request, provide a summary evaluation of the flow-induced vibration for the SG U-bend tubes based on the increase in feedwater flow and the increase in pressure difference between the primary system pressure (unchanged at 2250 psi) and the decreased steam pressure for the proposed power uprate.

25. In reference to Section 12.2.5, “Safety-Related Motor Operated Valves,” the licensee evaluated the effect of the proposed power uprate on the motor-operated valves program at IP3 for Generic Letter (GL) 89-10 and the 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 the response of GL 96-06 regarding overpressurization of isolated piping segment.

26. Provide a summary evaluation of the effect of the proposed power uprate on the design basis analysis for high energy line breaks, intermediate energy line breaks, jet impingement and pipe whip restraints.