November 5, 2004

Mr. Michael R. Kansler, President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING STRETCH POWER UPRATE, INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. MC3352)

Dear Mr. Kansler:

By letter dated June 3, 2004, Entergy Nuclear Operations, Inc. (ENO) submitted an application to increase the licensed thermal power level at Indian Point Nuclear Generating Unit No. 3 (IP3).

The Nuclear Regulatory Commission staff is reviewing the information provided in the 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 on November 3, 2004, the ENO staff indicated that a response to the RAI would be provided within 45 days.

If you should have any questions, please do not hesitate to call me at (301) 415-1457.

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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REQUEST FOR ADDITIONAL INFORMATION

REGARDING STRETCH POWER UPRATE (SPU)

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (IP3)

DOCKET NO. 50-286

By letter dated June 3, 2004 (ADAMS Accession No. ML041620506), Entergy Nuclear Operations (ENO or the licensee) submitted its application to increase the licensed thermal power level by 4.85% at IP3. The Nuclear Regulatory Commission (NRC) staff has the following questions regarding the information provided:

Reactor Systems

1. In Attachment III (application report) to the June 3, 2004, application, the licensee states the requirements of 10 CFR 50.68(b) apply to IP3 and remain valid for the upgraded fuel design. Explain when the licensing basis changed from 10 CFR 70.24 requirements to 10 CFR 50.68 requirements, and state the specific references by which the change was requested and approved.

2. Regarding the charging and volume control system (CVCS) malfunction re-analysis:

    a. The licensee assumes complete mixing of the diluted water injected through the cold leg with the active volumes in the RCS. Explain how a dilution front is addressed in the analysis for each plant mode and how a local power spike in the reactor core is precluded.

    b. The IP3 Final Safety Analysis Report states the reactor coolant system (RCS) volume assumed in the analysis was 8,630 ft^3 for Modes 1 and 2. However, the volume in the application report is 9,350 ft^3 for Modes 1 and 2. Provide the justification for the change in RCS volume used in the analyses. Was the methodology used for the SPU analysis consistent with the analysis of record?

3. The licensee states generic transient statepoints designed to bound IP3 at SPU conditions were used in the rod control cluster assembly (RCCA) Drop/Misoperation re-analysis. State the document which references these statepoints and demonstrate they are applicable to IP3.

4. Regarding the loss-of-normal feedwater (LONF) transient analysis:

    a. Currently, the turbine-driven auxiliary feedwater pump (TDAFWP) is not credited to mitigate this transient. In its SPU submittal, the licensee states the TDAFWP is valved out during normal operation. Therefore, although the TDAFWP is automatically actuated, this pump is not available to deliver flow to the steam generator until operator action is taken to align the TDAFWP. Provide a detailed description of the steps the operator takes to complete this action, and justify the operators are capable of performing this action in 10 minutes.

b. The analysis of record states the motor-driven auxiliary feedwater pumps are assumed to supply flow within 70 seconds of initiating signal. Explain why the new time of 60 seconds is stated in the application report, and show this is a conservative assumption.

5. In Section 7.4 of the application report, the licensee states rod internal pressure and clad fatigue criteria were met for the SPU condition. Provide the technical justification explaining how maintaining a vessel temperature of 572 °F will meet the rod internal pressure and clad fatigue criteria for the SPU operation. Also, provide the analytical basis that shows the clad fatigue criterion is met under SPU core conditions with a vessel average temperature of 572 °F.

6. The licensee used the LOFTTR2 computer code in its SPU steam generator tube rupture re-analysis. Demonstrate that the code is applicable for use at IP3 and that all conditions and limitations are met.

7. Regarding the complete loss-of-reactor-coolant flow transient analysis:

a. The licensee assumed an undervoltage trip time delay of 1.5 seconds. Explain the reason for the time delay and why this is a conservative assumption in the analysis.

b. The licensee assumed rod motion occurs at 1.6 seconds, following a 0.6 second underfrequency trip time delay. Explain the reason for the time delay and why this is a conservative assumption in the analysis.

8. Many cycle-specific parameters have been relocated to the core operating limits report (COLR), which was not submitted with the application. These include the Reactor Core Safety Limit (RCSL) figure, the values of the constants in the Over-temperature and Over-power ΔT functions, respectively (Notes 1 and 2 in Table 3.3.1-1), and the limit values of the pressurizer pressure, RCS average temperature, and RCS total flow rate (limiting condition for operation (LCO) 3.4.1).

Provide the RCSL figure and the values of the constants of the OTΔT and OPΔT functions, and the RCS flow, average temperature and pressure limit values in the COLR, or confirm that they are the same as those provided in the application report.

9. Section 6.3.1 of the application report describes the revised instrumentation ranges for RCS temperature measurement chosen for IP3 after implementation of the SPU to ensure proper operation of the OTΔT and OPΔT reactor trip functions over a realistic full power operating T_{avg} range of 562.0 °F to 572.0 °F as follows:

$520\ ^{\circ}F \le T_{cold} \le 640\ ^{\circ}F$
$540\ ^{\circ}F \le T_{avg} \le 615\ ^{\circ}F$
$520\ ^{\circ}F \le T_{hot} \le 640\ ^{\circ}F$

These revised instrumentation ranges are said to be derived from the instrumentation ranges for proper operation of the OTΔT and OPΔT functions over the entire range (T_{avg} from 549.0 °F to 572.0 °F) and a reduced, more realistic range (T_{avg} from 562.0 °F to 572.0 °F), respectively, of applicable full power operating RCS temperatures. Please provide a more detailed explanation of how these revised instrumentation ranges are derived.

11. Footnote 7 of Table 2.1-2 in the application report indicates that the RCS minimum measured flow of 364,700 gpm includes a 2.9% flow measurement uncertainty. Attachment I to the June 3 application regarding the proposed TS changes states that the SPU flow measurement uncertainty was calculated using the existing methodology described in WCAP-11397-P-A, and remains at the current value of 2.9%. Since WCAP-11397-P-A simply uses, rather than calculates, the RCS flow measurement uncertainty value, provide the calculation that shows the 2.9% RCS flow measurement uncertainty is applicable for use at IP3 under SPU conditions.

12. In Section 4.1.4 on the safety injection system/containment spray system of application report, the licensee states the high-head safety injection (HHSI) system was modified by permanently closing two cold leg branch lines, and throttling the HHSI system to provide higher cold leg and hot leg flows. What is the design function of the branch line? Why is it acceptable to defeat this function permanently? Provide a system diagram which depicts the branch lines that were closed. Discuss how this is administratively controlled and how the redundancy and independence of the system was preserved.

<u>Electrical</u>

1. Address and discuss the following points:

    a. Identify the nature and quantity of megavolt-ampheres reactive (MVAR) support necessary to maintain post-trip loads and minimum voltage levels.

    b. Identify the MVAR contributions that IP3 is credited for providing to the offsite power system or grid.

    c. After the power uprate, identify any changes in MVAR quantities associated with Items a. and b. above.

    d. Discuss any compensatory measures to adjust for any shortfalls in Item c. above.

    e. Evaluate the impact of any MVAR shortfall listed in Item d. above on the ability of the offsite power system to maintain minimum post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges with the transmission system operator.

2. The licensee stated that for Phase 1 of the stretch power uprate (1080 MWe), the isophase bus (IPB) bus duct is capable of operating within its ratings. In Phase 2, the main generator's output will increase to 1093.5 MWe. Increasing the generator output

to 1093.5 MWe, and operating the generator within the proposed reactive power limits, causes the IPB duct to operate slightly outside its rating. The load is only exceeded during extreme system grid conditions and can be permanently addressed with future Phase 2 modifications to the IPB. Describe in detail the Phase 2 modifications to the IPB duct.

Instrumentation and Controls

1. During the September 14, 2004, meeting between the licensee and the NRC staffs, the licensee stated that the IP3 RPS and ESFAS TSs allowable values (AVs) were calculated based on Instrument Society of America (ISA) Standard 67.04, part II, Method 3. However, in Attachment I, Section 2, "Proposed Changes," the application stated "a calculation method using ISA-RP67.04 Method 2."

   Clarify which method was used for each of the protective functions in the IP3 SPU application. Discuss the differences from methodology used for IP2.

2. Explain how the component test procedure acceptance criteria are determined and show that they do indeed provide adequate assurance that the channel AVs are suitably protected. Explain how this approach is compatible with the requirements of 10 CFR 50.36, which requires that the limiting safety system settings be specified in the TSs. Since channel performance is not assessed against the TS AVs unless some other criterion indicates that closer examination is warranted, those other criteria, which are not controlled by the TSs, can result in the TS criteria not being applied. Discuss the differences in methodology used for IP2 and 3.

3. Explain the difference in trip functions listed on Table 6.10-1, "SPU Summary of RTS/ESFAS Setpoint Calculations," for those in the IP2 SPU. IP2 has trip functions setpoint changes for "RCS flow low reactor trip," "SG [steam generator] level low-low reactor trip," "SG level high-feedwater isolation," and "Steamline pressure low SI/SL actuation." These changes were not listed on Table 6.10-1 of IP3. On the other hand, IP3 has trip functions setpoint changes for "Pressurizer pressure low reactor trip", and "Pressurizer pressure low SI initiation." These changes were not listed on Table 6.10-1 of IP2.

4. Provide the setpoint calculation documents for the following IP3 protection system trip functions listed in Table 6.10-1:

   a. Nuclear instrumentation system (NIS) power range reactor trip high setpoint function

   b. Overtemperature ΔT reactor trip and Overpower ΔT reactor trip functions

   c. Pressurizer pressure low reactor trip function

   d. Pressurizer pressure low safety injection initiation

   e. Steam flow in two steamlines-high (SI/SL actuation)

f. T_{avg} -low coincidence with high steam flow (SI/SL actuation)

5. Provide a copy of “IP3 Engineering Standard IES-3B, Instrument Loop Accuracy and Setpoint Calculation Methodology, Rev. 0," listed as Reference 2 in Section 6.10.5.

6. Provide a statement to clarify that no modification to the existing instrumentation and controls are required for the stretch power uprate except for certain RPS/ESFAS nominal trip setpoint and TSs allowable value changes and that the IP3 instrumentation and control systems will continue to perform their intended functions as required by plant license.

Mechanical Engineering

1. Section 3.1.3 of the application report states that the IP3 Model 44F SG design includes a primary-to-secondary pressure differential design limit of 1550 psid and this limit has been set to 1700 psid to minimize plant impact. This limit has also been used at IP2. Confirm how the set limit of the primary-to-secondary pressure differential is acceptable to be higher than the design basis.

2. In Table 3.1-1, the values of T_{steam} and P_{steam} at Low T_{avg} and T_{feed} for the present design do not match the values of Table 2.1-1 and Table 2.1-2. Explain why these values are different.

3. In Table 5.1-1, the numerical value of the stress intensity for the CRDM Housings is less than the allowable American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section III value of $3S_m$. However, the equation is written to indicate that the $3S_m$ value is exceeded. This error appears to be editorial. Confirm whether our observation is valid.

4. In Table 5.1-1, the stress intensity of the reactor vessel closure studs is very close to the allowable ASME Code allowable limit. Provide a summary of the calculation of the stress intensity and the cumulative usage factors (CUFs) for the closure studs.

5. Page 5.2-13 of the application states that the reactor pressure vessel internals were designed to meet the intent of Subsection NG of the ASME Code, Section III. It also states that a plant-specific stress report on the reactor pressure vessel internals was not required, and that the structural integrity of the internals has been ensured by analyses performed on both generic and plant-specific bases. Provide the calculated stresses at the uprated power level for components listed in Table 5.2-1 in comparison to the ASME Code allowed stress limits.

6. In Section 5.2.4.2, “Flow Induced Vibration (FIV),” the licensee indicated that, based on the analysis for the IP3 reactor internals, the response due to FIVs was extremely small and well within the allowable levels based on the high-cycle endurance limit for the materials. Provide a summary of the evaluation regarding the quantity of the response and the FIV analysis with respect to the fluid elastic instability, turbulent flow and vortex shedding and acoustic resonance. Also, provide the calculated vibration level and describe the allowable limit in your acceptance criteria.

7. In Table 5.3-2, the stress in the lower joint of the control rod drive mechanism (CRDM) canopy after the SPU will exceed the ASME Code allowable value of $3S_m$. The footnote states that the difference is insignificant. Provide the justification on how this issue was resolved.

8. Page 5.4-3 of the application states that the computer code WESTDYN is used to perform a system analysis of reactor coolant loop (RCL) piping. Confirm whether the WESTDYN code has been reviewed and approved by the NRC. If not, provide justification for using this code.

9. Section 5.5.2.4 describes the acceptance criteria for the reactor coolant pump motor loading. It states that the temperature rise of the motor while driving the pump continuously under hot-loop conditions with an ambient temperature of 120 °F must be in accordance with National Electrical Manufacturer's Association (NEMA) Standard MG1-20.40-1963. However, Page 5.5-6 states that the temperature rise of the motor while driving the pump continuously under hot-loop conditions with an ambient temperature of 130 °F will meet NEMA Standard MG1-20.40-1963. Explain the reason for the difference between these two temperatures.

10. In Table 5.5-1, the T_{cold} value given for High T_{avg} does not match the value given in Table 2.1-2. Explain this discrepancy.

11. With regard to Section 5.6.1 of the application, provide an evaluation for the effect of FIV on the SG steam dryer, and dryer supports with respect to the fluid-elastic instability, acoustic loads, turbulence and vortex shedding due to the increased steam flow for the power uprate.

12. Section 5.9.1 of the application states that IP3 has a Model 44F pressurizer. However, a Model D series 84 pressurizer was used in the design-basis analysis for IP3. Model D series 84 pressurizer has the same dimensions and materials as the Model 44F. Discuss the applicability of using the base analysis of a Model D pressurizer to the IP3 Model 44F pressurizer considering the structural characteristics between these two models including supports and structural natural frequencies.

<u>Reactor Vessel, Pressurizer, and Steam Generator Structural Integrity</u>

1. In its June 16, 2004, response to a request for additional information on the IP2 SPU, the licensee provided information, in part, regarding pressure vessel materials (PVM). In PVM question no. 1 (PVM-1), the NRC staff had requested that the licensee evaluate the impact of surveillance data on the Charpy Upper Shelf Energy (USE) of the IP2 reactor vessel (RV) beltline materials. Although the licensee will be responding to the same question for IP3, the following additional information is necessary:

    a. Section 5.1.1.2 of the application report indicates that the minimum inlet temperature decreased from 542.2 °F to 517.2 °F for SPU conditions. In response to the issues in PVM-1, the licensee needs to include an evaluation of the impact of the lower inlet temperature on the predicted Charpy USE at end of license (27.1 effective full power years at SPU conditions).

b. PVM-1 also requested the licensee to justify the use of a smaller flaw size for the RV outlet-nozzle-to-shell region. Table 5.9-4 of the IP3 application report indicates that smaller flaw sizes than those specified in Appendix G of Section XI of the ASME Code were used for the steam generator tubesheet and shell junction, steam outlet nozzle, and feedwater nozzle. Provide a justification for using the smaller size flaw or provide an analysis using the flaw sizes and margins described in Appendix G of the Section XI of the ASME Code.

2. PVM question 2 (PVM-2) should be replaced with the following RAI:

Table 5.1.3 of the application report identifies the RT_{PTS} values for the IP3 RV beltline materials at the end of license at SPU conditions. Section 5.1.2.2 indicates that the Pressure-Temperature (P-T) Limit Curves will be reduced by 0.7 EFPY as a result of SPU conditions. The material with the highest RT_{PTS} value at end of license at SPU conditions is Lower Shell Plate B2803-3 with an RT_{PTS} value of 262 °F, using surveillance data, and 268 °F, based on its chemical composition. This material also has the highest adjusted reference temperature (ART) in the P-T Limit Curve evaluation. The RT_{PTS} and ART values were determined using the methodology contained in Regulatory Guide (RG) 1.99, Revision 2. This guide indicates that the procedures are valid for nominal irradiation temperature of 550 °F and irradiation below 525 °F should be considered to produce greater embrittlement.. Since the licensee proposes to reduce the inlet temperature to 517.2 °F, the licensee must determine the impact that operating with inlet temperatures below 525 °F has on the RT_{PTS} and ART values.

Provide the surveillance data for Plate B2803-3 and include a credibility evaluation of the surveillance data in accordance with RG 1.99, Revision 2. Identify the mean inlet temperature and peak RV neutron fluence for each cycle of RV operation (include data from cycles prior to SPU and projected for post SPU conditions). Identify the mean inlet temperature for each surveillance capsule. Provide an evaluation of the impact of the lower inlet temperature on the predicted RT_{PTS} and ART values for this plate and all other materials in the IP3 RV beltline. Describe how this evaluation impacts the P-T Limit Curves. Identify how the inlet temperature will be monitored during SPU conditions to confirm that the projected RT_{PTS} and ART values remain valid for operation at SPU conditions.

In addition, if SPU conditions result in a change in the period of applicability for the P-T Limit Curves, the P-T Limit Curves must be submitted for staff review as part of the TS amendment process.

3. Section 5.1.2.1 of the application report indicates that a fifth surveillance capsule must be withdrawn to satisfy regulatory requirements and that the withdrawal schedule is presented in Table 5.1-2. This table provides options for withdrawal of the fifth capsule; but, does not specify the date of capsule withdrawal. Appendix H to 10 CFR Part 50 requires that the withdrawal schedule be submitted and approved prior to implementation. Provide the date for withdrawal of the fifth capsule and describe how it complies with regulatory requirements.

Piping

1. In Section 5.10.4, "Change in PWSCC [Primary Water Stress Corrosion Cracking] Susceptibility of RVHP [Reactor Vessel Head Penetrations]," the licensee uses the RV upper head best-estimate mean fluid maximum service temperature for the purpose of determining the change in PWSCC susceptibility. The NRC staff does not find this calculation using the mean fluid temperature conservative. The staff finds that using the RV upper head maximum temperature to determine the maximum change in the PWSCC susceptibility value would be appropriate and conservative. Therefore, the licensee should update Table 5.10-1 and perform the change in PWSCC susceptibility calculation with appropriate data inputs.

2. In Section 5.10.6, "Conclusions," the licensee states "The increase in PWSCC susceptibilities of Alloy 600 RVHP and hot-leg nozzle weld locations (22 and 9 percent) indicated above is not considered significant since the absolute susceptibility of these locations is estimated to be very low ($\sim 10^{-11}$)." The staff finds the $\sim 10^{-11}$ value to be inconsistent with industry inspection results and analysis performed in Material Reliability Program (MRP) Reports MRP-105, "Materials Reliability Program Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking" and MRP-110, "Materials Reliability Program Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants."

   a. Provide the basis for this estimated absolute susceptibility value.

   b. What other actions will the licensee take to address the increase in PWSCC susceptibilities of Alloy 600 RVHP and hot-leg nozzle weld locations of 22% and 9%, respectfully?

   c. What augmented inspections will be performed by the licensee on Alloy 182/82 welds in the reactor coolant pressure boundary hot leg?

Flow-Accelerated Corrosion Program

1. Section 10.3, "Flow-Accelerated Corrosion [FAC] Program," states that the CHECWORKS™ program is used to predict erosion rates for several large-bore high-energy piping systems.

   a. Describe the criteria used in for selecting components for modeling using the CHECWORKS™ Program.

   b. Describe the criteria for repair or replacement of components that become damaged as a result of FAC.

   c. For the five components most susceptible to FAC, provide numerical data that show changes in: (1) velocity, (2) temperature, and (3) predicted wear rate that result from the SPU.

2. Section 10.3 also states that the IP3 Small Bore and Augmented Monitoring Program is used to address piping that has not been modeled using CHECWORKS™ program.

   a. Describe the Small Bore and Augmented Monitoring Program in more detail. Describe the criteria (in addition to piping diameter requirements) that determines which small bore lines are included in the Program.

   b. Describe the criteria used to include non-small bore piping into the Small Bore and Augmented Monitoring Program instead of the CHECWORKS™ program.

   c. Describe how the Small Bore and Augmented Monitoring Program predicts erosion rates in small bore lines.

Protective Coatings Program

1. The NRC staff notes that the application did not include a description of the Protective Coatings Program at IP3.

   a. Describe the Protective Coatings Program for IP3.

   b. Discuss in general terms how the SPU affects the Protective Coatings Program.

   c. Discuss how the qualification of the Service Level 1 coatings are impacted by SPU temperature and pressure conditions.

   d. Discuss whether the qualification parameters (e.g., temperature, pressure, etc.) for your Service Level 1 coatings will continue to be bounded by SPU design-basis accident (DBA) conditions.

   e. Describe the actions that will be taken if the qualification of Service Level 1 coatings are not bounded by the SPU/DBA conditions, since coating failure could threaten performance of the ECCS sump after a LOCA.

Steam Generator Structural Integrity Evaluation

1. The conclusions for mechanical plugs in Section 5.6.4 states that "... both the long and short mechanical plug designs satisfy all applicable stress and retention acceptance criteria at the SPU operating conditions with up to 10-percent SGTP [steam generator tube plugging]." The results subsection states that "The plug meets the Class 1 fatigue exemption requirements per N-415.1 of the ASME Code ..."

   a. Provide a table (similar to Table 5.6-2 for the primary and secondary side components) which summarizes the load conditions, stress categories, ASME allowables, and all applicable stress- and fatigue-related calculation results that support your conclusions for the mechanical plugs. Show the calculation results which indicate that ASME Code allowables were met.

   b. Provide calculation results which show that the mechanical plugs are qualified for the SPU condition with up to 10% tube plugging.

c. Provide the basis and calculation results (if any) for satisfying the ASME Class 1 fatigue exemption requirements.

2. The conclusions for shop welded plugs in Section 5.6.4 states that "[a]ll primary stresses are satisfied for the weld between the weld plug and the tubesheet cladding," and "[t]he overall maximum primary-plus-secondary stresses for the enveloping transient case of 'loss-of-load' was determined to be acceptable," and "[i]t was determined that the fatigue exemption rules were met, and therefore, fatigue conditions are acceptable."

   a. Provide a table (similar to Table 5.6-2 for the primary and secondary side components) that summarizes the load conditions, stress categories, ASME Code allowables, and all applicable stress- and fatigue-related calculation results that support your conclusions for the shop weld plugs. Show the calculation results which indicate that ASME Code allowables were met.

   b. Provide the basis and calculation results (if any) for satisfying the ASME Code fatigue exemption requirements.