December 30, 2003

MEMORANDUM TO: Darrell J. Roberts, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Richard B. Ennis, Senior Project Manager, Section 2 ***/RA/***
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION, DRAFT REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MC0761)

The attached draft request for information (RAI) was transmitted on December 18, 2003, to Ms. Ronda Daflucas of Entergy (the licensee). This draft RAI was transmitted to facilitate an upcoming conference call in order to clarify the licensee's amendment request dated September 10, 2003, as supplemented by letters dated on October 1, 2003, and two dated October 28, 2003. The proposed license amendment would allow an increase in the maximum authorized power level from 1593 megawatts thermal (MWT) to 1912 MWT.

This memorandum and the attachment do not convey or represent an NRC staff position regarding the licensee's request.

Docket No. 50-271

Attachment: Draft RAI

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DRAFT REQUEST FOR ADDITIONAL INFORAMTION

REGARDING PROPOSED LICENSE AMENDMENT

EXTENDED POWER UPRATE

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

By letter dated September 10, 2003, as supplemented by letters dated on October 1, 2003, and two dated October 28, 2003, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a proposed license amendment for Vermont Yankee Nuclear Power Station (VYNPS). The proposed amendment would allow an increase in the maximum authorized power level from 1593 megawatts thermal (MWT) to 1912 MWT.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information the licensee provided that supports the proposed amendment and would like to discuss the following issues to clarify the submittals.

ATTACHMENT

**Electrical and Instrumentation and Controls Branch (EEIB)**

Instrumentation and Controls Section (EEIB-A)
Reviewer: Hukam Garg

1. Based on Section 5.1 of the staff's Safety Evaluation of General Electric Nuclear Energy (GENE) Licening Topical Report (LTR) NEDC-33004P, "Constant Pressure Power Uprate," (CPPU) dated March 31, 2003, the staff requested that the plant-specific submittal address all CPPU-related changes to instrumentation and controls, such as scaling changes, changes to upgrade obsolescent instruments, and changes to the control philosophy. The licensee has not provided this information.

**Electrical and Instrumentation and Controls Branch (EEIB)**

<u>Electrical Engineering Section (EEIB-B)</u>
Reviewer: Narinder Trehan

1. Provide the results of the additional analysis referenced in Section 10.3.1 of Attachment 6 of your submittal dated September 10, 2003, for the effect of the EPU on the environmental qualification of electrical equipment in harsh environments located inside and outside the containment.

2. Provide in detail information regarding the extensive modifications to the main generator rewind/upgrade, generator hydrogen coolers, and isolation phase bus duct coolers.

3. Provide the evaluation, referenced in Section 6.1.2 of Attachment 6 of your submittal dated September 10, 2003, of the operation of the Condensate and Reactor Feedwater Pump motors at higher summer temperatures at the power uprated condition.

4. Address the compensatory measures that the licensee would take to compensate for the depletion of the nuclear unit mega-volt-amperes reactive (MVAR) capability on a grid-wide basis.

**Materials and Chemical Engineering Branch (EMCB)**

Vessels and Internals Integrity and Welding Section (EMCB-A)
Reviewer: Barry Elliot

1. Section 3.2.2, “Reactor Vessel Structural Evaluation,” of Attachment 6 to your submittal dated September 10, 2003, indicates a fracture mechanics analysis was used in conjunction with inner surface exams and cycle counting to assure potential crack growth is smaller in relation to the ASME XI limits for the feedwater (FW) nozzle blend radius location. The Ultrasonic Testing (UT) inspection of the inner surface of the FW nozzles is based on a BWROG report GE-NE-523-A71-0594, Revision 1, August 1999, that was approved by the NRC in a letter dated March 10, 2000. The fracture mechanics analysis evaluates crack growth for conservative design transients. The conservative design transients used in the fracture mechanics evaluation conservatively bound changes under CPPU conditions.

   Identify the design transients used in the fracture mechanics analysis, compare these transients with those assumed under CPPU conditions and explain why CPPU conditions do not impact the fracture mechanics analysis.

2. Section 10.7, “Plant Life,” of Attachment 6 to your submittal dated September 10, 2003, identifies irradiation-assisted stress-corrosion cracking (IASCC) as a degradation mechanism influenced by increases in neutron fluence. This section indicates that the current inspection strategy for reactor internal components is expected to be adequate to manage any potential effects of CPPU. Note 1 in Matrix 1 of Section 2.1 of RS-001, Revision 0 indicates guidance on the neutron irradiation-related threshold for inspection for IASCC in BWRs is in Boiling Water Reactor Vessel and Internals Program (BWRVIP) report BWRVIP-26. The “Final License Renewal SER for BWRVIP-26,” dated December 7, 2000, states that the threshold fluence level for IASCC is $5 \times 10^{20}$ n/cm2 (E > 1 MeV). Identify the vessel internal components whose fluence at the end of period of operation with CPPU conditions will exceed the threshold level and become susceptible to cracking due to IASCC. For each vessel internals component that exceeds the IASCC threshold, either provide an analysis that demonstrates failure of the component will not result in the loss of the intended function of the reactor internals or identify the inspection program to be utilized to manage IASCC of the component. Identify the scope, sample size, inspection method, frequency of examination and acceptance criteria for the inspection programs.

**Materials and Chemical Engineering Branch (EMCB)**

Piping Integrity and Non-Destructive Examination Section (EMCB-B)
Reviewer: Robert Davis

1. Section 3.5.1 of Attachment 4 of your submittal dated September 10, 2003, provides the results of the structural evaluation of the reactor coolant pressure boundary (RCPB) piping. Provide the basis for the disposition of the first system listed in this section.

2. Identify the materials of construction for the Reactor Recirculation System piping and discuss the effect of the requested EPU on the material. If other than type “A” (per NUREG 0313) material exist, discuss augmented inspection programs and discuss the adequacy of augmented inspection programs in light of the EPU.

3. Section XI of the American Society of Mechanical Engineers (ASME) Code allows flaws to be left in service after a proper evaluation of the flaws is performed in accordance with the ASME, Section XI rules. Indicate whether such flaws exist in the Reactor Recirculation System piping and evaluate the effect of the EPU on the flaws.

4. Discuss flaw mitigation steps that have been taken for the RCPB piping and discuss changes, if any, that will be made to the mitigation process as a result of the EPU.

**Materials and Chemical Engineering Branch (EMCB)**

Steam Generator Integrity and Chemical Engineering Section (EMCB-C)
Reviewer: Krzysztof Parczewski

1. In Section 4.2.6 of Attachment 6 to the submittal dated September 10, 2003, the licensee states that the debris loading on the suction strainers and the methodology used to calculate available Emergency Core Cooling System (ECCS) net positive suction head (NPSH) for CPPU are the same as the pre-CPPU conditions. What assumptions are used with respect to failure of protective coatings and organic materials for the post-accident performance of the ECCS (pre-CPPU and post-CPPU)?

2. In Section 10.7 of Attachment 6 to the submittal dated September 10, 2003, the licensee addresses the Flow-Accelerated Corrosion (FAC) program for VYNPS. The program consists of inspecting selected components and subsequently using the inspection results to qualify all the FAC susceptible components for further service. In order to evaluate the licensee's program, the staff requires the following additional information:

    a. In the FAC program, what are the criteria for selecting components for inspection after the EPU?

    b. What are the changes in the predicted wear rates after the EPU in the Main Steam Drains, Moisture Separator Drains, and the Turbine Cross Around System piping?

    c. What are the changes of velocity and temperature of the feedwater caused by the EPU?

3. In Section 3.11 of Attachment 6 to the submittal dated September 10, 2003, the licensee addresses the Reactor Water Cleanup System (RWCS) evaluation. The staff requires the following additional information:

    a. By how much does the temperature in the RWCS decrease after the EPU?

    b. What is the expected increase of iron input to the reactor caused by a higher feedwater flow?

    c. In the submittal, the licensee stated that its review of the RWCS functional capability has indicated that during EPU the system can adequately perform with the original RWCS system flow. Provide a basis for this conclusion.

**Mechanical and Civil Engineering Branch (EMEB)**

Civil and Engineering Mechanics Section (EMEB-B)
Reviewer: Cheng-Ih (John) Wu

1. Sections 3.5 and 4.1.2, of Attachment 6 to the submittal dated September 10, 2003, provide a discussion of the evaluation of piping systems attached to the torus shell, vent penetrations, pumps, and valves, that are affected by increased torus temperature and changes in loss-of-coolant accident (LOCA) dynamic loads (pool swell, condensation oscillation, and chugging) and increased temperature and flow in the main steam and feedwater systems due to the proposed power uprate. Identify supports and piping systems affected by required modifications stated in Attachment 3 of the submittal, as a result of the proposed EPU.

2. Section 3.5.2, of Attachment 6 to the submittal dated September 10, 2003, provides a summary addressing your evaluation of the effects of the proposed power uprate on the BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorages. Also, provide the calculated maximum stresses and fatigue usage factors for the most critical BOP piping systems, the allowable limits, the code of record and code edition used for the power uprate conditions. If different from the code of record, justify and reconcile the differences.

3. On page 3-15 of Attachment 6 to the submittal dated September 10, 2003, it states that a qualitative evaluation was performed that identified preliminary modifications and inspections to enhance the structural integrity of the steam dryer at CPPU conditions, and that a quantitative evaluation to identify dryer components susceptible to failure at CPPU conditions is being performed. The licensee should describe those evaluations and their results, and the schedule for implementing identified modifications. The licensee should also describe its evaluation of the recommendations in General Electric (GE) Service Information Letter (SIL) No. 644, Revision 1, “BWR Steam Dryer Integrity,” and its commitments regarding implementation of those recommendations.

4. On page 3-23 of Attachment 6 to the submittal dated September 10, 2003, it states that the increase in steam flow rate under CPPU conditions will assist in the closure of the Main Steam Isolation Valves (MSIVs) at VYNPS. The licensee indicates that the self-compensating feature of the hydraulic control valve will maintain the closing time with little deviation despite the flow change. The licensee should describe the MSIVs, the design feature that will ensure that MSIV closure time is not reduced below the stroke-time limit, and any testing or operating experience from plant-specific or generic sources that supports its determination that closure time will remain within the allowable limits.

5. On page 4-6 of Attachment 6 to the submittal dated September 10, 2003, the licensee indicates that it evaluated the Generic Letter (GL) 89-10 motor-operated valves (MOVs) at VYNPS for the effects of the CPPU, including those related to pressure locking and thermal binding per GL 95-07. The licensee reports that there were no changes to the design functional requirements of the MOVs. The licensee states that it did identify minor process fluid condition changes and increased ambient room temperatures for some MOVs. The licensee indicates that it will evaluate the affected MOVs through

MOV program calculation updates with any resulting changes in current MOV settings implemented prior to CPPU operation. On page 4-7, the licensee reports that air-operated valves (AOVs) were reviewed to identify AOVs potentially affected by CPPU conditions. The licensee states that evaluation of affected AOVs may identify setting changes or modifications that will be accomplished prior to CPPU implementation. The licensee should describe the status, methodologies, and results of those MOV and AOV evaluations, and describe any planned setting changes or modifications. The licensee should also clarify: (1) the effect of the power uprate on the potential for thermal binding or pressure locking, such as caused by temperature increases, on the scope of power-operated valves under GL 95-07 or the performance of those valves; and (2) any modifications or procedure changes necessary as a result of the power uprate to preclude thermal binding and pressure locking. Finally, the licensee should describe its plans to incorporate the results of its evaluation of MOV performance under CPPU conditions into its long-term program to periodically verify the design-basis capability of safety-related MOVs in response to GL 96-05.

6. Beginning on page 4-7 of Attachment 6 to the submittal dated September 10, 2003, the licensee indicates that the performance of the ECCS at VYNPS will remain acceptable under CPPU conditions. In addition to this general discussion, the licensee should describe its evaluation of the effects of the CPPU on the performance of safety-related pumps; and any modifications or procedural changes related to safety-related pumps that might be necessary to accommodate normal plant operations under CPPU conditions or to support performance of their safety functions during design-basis events subsequent to implementation of the CPPU.

7. On page 6-3 of Attachment 6 to the submittal dated September 10, 2003, the licensee notes that load changes on the DC power distribution system could include DC MOV load increases. The licensee should describe the potential DC MOV load increases resulting from the CPPU, and the impact of those load increases on DC MOV performance.

8. On page 10-2 of Attachment 6 to the submittal dated September 10, 2003, the licensee states that operation at CPPU conditions will result in a slight increase in downcomer subcooling that may lead to increased flow rates for liquid line breaks. The licensee should describe the operation of the applicable pumps or valves under those increased flow rates, and any adverse performance effects.

**Emergency Preparedness and Plant Support Branch (IEPB)**

Quality and Maintenance Section (IEPB-A)
Reviewers: Robert Pettis, Kevin Coyne

1. Supplement 3, dated October 28, 2003, provided an update to Attachment 3 of the September 10, 2003, submittal which addressed the licensee's EPU testing and modification plans. Attachment 3, page 21, states for STP 23 (feedwater system) and STP 24 (bypass valves), that testing is planned for CPPU. However the "Evaluation/Justification for Not Performing Test" column of the table states that testing is not required. Please provide clarification.

**Emergency Preparedness and Plant Support Branch (IEPB)**

Emergency Preparedness and Health Physics Section (IEPB-B)
Reviewer: Roger Pedersen

1. Section 8.6, “Normal Operation Off-Site Doses,” of Attachment 6 to the submittal dated September 10, 2003, states in the last paragraph, that the increased N-16 source at the turbine is due to lower decay times in transient, due to the higher steam flow rate and gives an expected increase of 26%. This percentage is then included in a maximum site boundary dose from all sources of 18.6 mrem. Provide a breakdown of this overall dose number. List all dose pathway components and describe the calculation method used, including all assumptions. Provide the present nominal value for the skyshine external dose component (before EPU) and the estimated value following EPU and identify the dose receptor for this skyshine component (i.e., is the dose receptor a member of the public located offsite (and therefore subject to the dose limits of 40 CFR 190) or a member of the public working onsite (subject to the dose limits of 10 CFR 20.1301)).

2. Section 6.3.3, “Radiation Levels,” of Attachment 6 to the submittal dated September 10, 2003, states that the normal radiation levels around the spent fuel pool (SFP) may increase slightly, primarily during fuel handling operations. Explain the reason for and the magnitude of these postulated increases in dose rate levels in the area of the SFP. Verify that these postulated dose rate increases will be bounded by the current radiation zone designations in the SFP area. If this postulated dose rate increase is due to higher activation of spent fuel assemblies, discuss any effects that the storage of these spent fuel assemblies in the SFP may have on dose rates in accessible areas adjacent to the sides or bottom of the SFP.

3. Section 8.3, “Radiation Sources in the Reactor Core,” of Attachment 6 to the submittal dated September 10, 2003, states that access to vital areas needed for accident mitigation have been demonstrated to be less than 5 rem TEDE. Provide a list of vital areas requiring post-accident occupancy, including the plant’s Technical Support Center, per NUREG-0737, Item II.B.2. For each of these vital areas, provide the calculated pre-uprate and post-uprate mission doses to an operator performing vital tasks following a LOCA. Specify the source term assumptions (e.g., core activity release timing assumed) in the post-CPPU/post-Alternative Source Term (AST) calculations.

4. Section 8.4.2, “Activated Corrosion Products and Fission Products” of Attachment 6 to the submittal dated September 10, 2003, states that there may be an increase in the activated corrosion product production, but does not quantify the expected increase in dose rates from the increase in activated corrosion products. Provide the following information: 1) verify that there is an expected increase in activated wear products as well as corrosion products; 2) what plant areas will be affected by the increase in production, transport and deposition of activated corrosion and wear products (i.e., areas where activated corrosion and wear products in systems are the major dose contributor); 3) what are the expected magnitudes of the dose rate increases associated with this impact; 4) provide the technical basis for the expected increase; and 5) what affects this will have on occupancy levels in the affected areas.

5. In Section 8.5, “Radiation Levels”, of Attachment 6 to the submittal dated September 10, 2003, the statement is made that the original designs for most plants are sufficiently conservative to compensate for increasing radiation levels from power increases. It goes on to state that “the normal operating radiation levels specified for CLTP conditions were evaluated to increase in proportion to the increase in thermal power.” This linear proportionality assumption may be valid for areas where the major source of radiation is the reactor core. However, this is not the case for much of the auxiliary buildings and balance of plant spaces. As noted in question 1 above, N-16 radiation at the turbine increases exponentially with decreased decay time, not linearly with the power increase. The higher rate of steam flow also reduces the hold-up time of the condensate in the condenser hot-well. Therefore, there should be increased N-16 in the condensate bearing systems from both an higher rate of input to the condenser and a reduced decay time. Provide the calculated increase in dose rates around the condensate system. Verify that the expected increase does not create new radiation, or high radiation, areas. Verify that the current plant shielding has sufficient design margin, and that the power increase will not affect the plant radiation zoning.

6. With respect to the 2nd sentence on page 8-5 of Attachment 4 to the submittal dated September 10, 2003, provide the specific locations of these areas where higher dose rates are predicted, give the reasons for the expected additional increase in radiation levels in these areas, state the percentage increase in dose rates expected, and state what measures will be put in place in these areas to ensure that dose to plant personnel is maintained ALARA.

7. In Section 8.2, “Gaseous Waste Management,” of Attachment 6 to the submittal dated September 10, 2003, you state that, “the radiological release rate is administratively controlled to remain within existing limits, and is a function of fuel cladding performance,...” and several other factors. Aside from limiting power (to the point of shutting down the plant, assuming gross fuel leakers, etc.), how can an operator administratively control gaseous effluents from the main condenser offgas during plant operation?

**Reactor Operations Branch (IROB)**

<u>Operator Licensing and Human Performance Section (IROB-B)</u>
Reviewers:  Richard Eckenrode, James Bongarra

1. With regard to operator responses, as described in Section 10.6 of Attachment 6 to the submittal dated September 10, 2003, the submittal includes a new task to be incorporated into plant procedures.  This task is described as “closing, from the Control Room, a normally open torus vent” in response to “a fire in the reactor building Appendix R event.”  Please describe the manual actions required to accomplish this task, including the indications required to recognize that the actions are necessary, the procedural steps involved in the actions, the time available for taking the actions, and the indications of successful completion.

2. The submittal states in regard to operator responses, as described in Section 10.6 of Attachment 6 to the submittal dated September 10, 2003, “the time available for some operator actions is reduced by small increments.”  Please provide both the bases for these reduced allowable action times and demonstration that the most time-limited actions can be accomplished by all operating crews.  Additionally, please describe the consequences of failure to meet the stated time limits.  In particular, the response should address the following Key Operator Actions from Table 10-5:  IABASE, IOPSLMCF, OPMSIVBP, and VROPERROR3.

**License Renewal and Environmental Impacts Branch (RLEP)**

Environmental Section (RLEP-C)
Reviewer: Stacey Imboden

1. Due to the EPU, there will be an increase in current across the transmission lines. Discuss the electric shock hazards associated with the increased current. Were the transmission lines designed and constructed in accordance with the applicable shock prevention provisions of the National Electric Safety Code?

2. What is the expected increase in water temperature at the discharge point due to the EPU? Approximately how far from the discharge will this temperature gradient spread out - will it dissipate immediately due to the mixing in the Connecticut River?

3. Is there any critical habitat in the vicinity of the river discharge? What organisms are in the vicinity of the discharge and how will they be affected?

4. Are there any aquatic species that could be caught in the intake structure? Are any of these Federally or State listed? Does Entergy have any protective measures to prevent aquatic species from entering the intake area?

5. How many full time employees and contractors work at VYNPS? Will the EPU affect the size of the labor force? Will the EPU have an affect on the labor force required for future outages? How many additional people are required for current outages?

6. Is Entergy a major employer in the community? Is Entergy a major contributor to the local tax base? What affect will the EPU have on the local tax base?

7. What is the volume of solid and liquid low-level radioactive waste (LLW ) currently generated (in calendar year 2002) at VYNPS? What is the average annual amount of solid and liquid LLW generated at VYNPS?

8. Due to the EPU, what is the increase in on-site occupational dose? What will be done to limit the increase?

9. Discuss the effect of skyshine on direct radiation doses offsite. How is whole body dose monitored at VYNPS? What is the highest annual offsite dose due to skyshine? How will the EPU affect dose due to skyshine? How will dose due to skyshine be monitored?

10. What is the uranium-235 enrichment value (weight percent of uranium-235) for fuel used for the EPU? What is the expected fuel burnup (in megawatt days per metric ton of uranium (MWd/MTU)) for the EPU?

**Plant Systems Branch (SPLB)**

Balance of Plant Section (SPLB-A)
Reviewer: Devender Reddy

Questions to be provided later.

**Plant Systems Branch (SPLB)**

Fire Protection Engineering and Special Projects Section (SPLB-B)
Reviewer: Ray Gallucci

1. In NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," Attachment 2 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that "... power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's application should confirm that these elements are not impacted by the extended power uprate ..." The reviewer notes that Section 6.7, "Fire Protection," of Attachment 6 to the submittal dated September 10, 2003, addresses only items (2) through (5) above. At a minimum, please provide a statement to address item (1), no effect upon "administrative controls," and a statement confirming no "increase in the potential for a radiological release resulting from a fire."

2. In RS-001, Revision 0, Attachment 2 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that "... where licensees rely on less than full capability systems for fire events ..., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability ... The licensee should identify the impact of the power uprate on the plant's post-fire safe shutdown procedures." The reviewer notes that Section 6.7, "Fire Protection," of Attachment 6 to the submittal dated September 10, 2003, addresses all but the following item above - "no adverse consequences on the reactor pressure vessel integrity or the attached piping" (the Application does address the effect on containment pressure and temperature). At a minimum, please provide a statement to address this item.

3. Section 6.7.1, "10 CFR 50 Appendix R Fire Event," of Attachment 6 to the submittal dated September 10, 2003, discusses an evaluation performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming CPPU conditions. The submittal states that the results of the Appendix R evaluation for CPPU provided in Table 6-5 demonstrate that fuel cladding integrity and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions." Upon reviewing Table 6-5 ("VYNPS Appendix R Fire Event Evaluation Results"), the reviewer was able to find references to only two of the values provided, namely the drywell design pressure of 56 psig and the containment structure design limit of 281°F for suppression pool bulk temperature, both from Table 4-1. The reviewer was also able to confirm Notes 5, "NPSH demonstrated adequate" (Section 4.2), and 6, "Overpressure credit required" (Section 4.2.6). Please provide

references, including appropriate extracts from the UFSAR, Appendix R evaluation, etc., for all remaining notes and values in Table 6-5.

**Probabilistic Safety Assessment Branch (SPSB)**

Safety Programs Section (SPSB-A)
Reviewer: Martin Stutzke

1. Attachment 6, Section 10.5, of the submittal dated September 10, 2003, states that the core-damage frequency (CDF) will increase from 7.77E-06/y to 8.10E-06/y as a result of the EPU. Section 10.5.7 states that the probabilistic safety assessment (PSA) model used for the analysis is VY02 Revision 6, which was completed in July 2003. In May 2003, the NRC conducted a benchmarking exercise of its Significant Determination Process (SDP) Phase 2 model by comparing its results to the licensee's PSA model. During the benchmarking exercise, the VYNPS PSA model was identified as Revision 3 (4/30/03), with a CDF due to internal initiators and internal floods of 4.89E-06/y. Please explain what changes were made to the VYNPS PSA model and why the CDF apparently increased during the May-July 2003 time frame.

2. Attachment 6, Section 10.5.7, of the submittal dated September 10, 2003, states that an industry peer review of the PSA was performed in November 2000. Please provide the Category A and B review findings, and explain how and when each finding was resolved.

3. Attachment 6, Section 3.1, of the submittal dated September 10, 2003, indicates that flow-induced vibration (FIV) may cause an inadvertent safety/relief valve (SRV) opening and a stuck-open SRV. However, Section 10.5.1 indicates that no effect on loss of coolant accident (LOCA) frequencies due to the EPU were postulated. Please resolve this apparent contradiction.

4. Attachment 6, Section 7.4.2, of the submittal dated September 10, 2003, states that a reactor recirculation system runback modification will be installed to avoid a plant trip on loss of a condensate pump or reactor feedwater pump (RFP). How has this modification been addressed in the PSA? Could malfunction of the runback circuitry cause a total loss of feedwater? If so, please describe how the total loss of feedwater initiating event frequency been modified.

5. Attachment 6, Section 10.5.4, of the submittal dated September 10, 2003, states that the EPU plant, including ARTS/MELLLA, has an additional spring safety valve (SSV) that provides additional relief capacity for the limiting ATWS transient. This section also states that the EPU configuration is "more than adequate with one SRV [safety relief valve] OOS [out-of-service]." However, Table 10-3 indicates almost the same CDF for Accident Class IVL (ATWS sequences where core damage occurs due to overpressure failure of the Reactor Coolant System) for both the current plant and the EPU plant. Please resolve this apparent contradiction.

6. Attachment 6, Section 6.1.1, of the submittal dated September 10, 2003, states that a grid stability study "is being performed." Please provide the frequencies of loss-of-offsite-power (LOOP) events due to plant-centered causes, grid-related causes, and weather-related causes, and describe how these values were developed (e.g., which data sources were consulted, etc.). Also describe how the probability of non-recovery

from LOOP events is calculated. The NRC staff notes that recent events within the U.S. suggest that the durations of LOOP events may be significantly longer than the past.

7. Attachment 6, Section 10.5.3, of the submittal dated September 10, 2003, states that a new operator action will be incorporated into the plant procedures to satisfy certain aspects of fire (Appendix R) and station blackout (SBO) evaluations. The new action requires the operators to close the normally open torus vent valve in order to maintain ECCS net positive suction head (NPSH) when the residual heat removal (RHR) system is operating in the containment spray system (CSS) mode. Section 10.5.3 concludes that since the PSA credits torus cooling (RHR operating in the suppression pool cooling - SPC - mode), this action has no direct applicability. Please describe the circumstances (scenarios, procedural symptoms, etc.) under which the operator is directed to close the torus vent valve. Also describe the circumstances under which the operator is directed to re-open the torus vent valve. Justify not defining new human failure events address improper control of the torus vent valve.

8. Attachment 6, Section 10.5.3, of the submittal dated September 10, 2003, describes the screening process used to identify which human error probabilities (HEPs) required adjustment to account for the shortened available times due to the EPU. Table 10-5 lists 41 operator actions whose HEPs were adjusted. Please provide a complete list of post-initiator operator actions included in the VYNPS PSA model, including their HEPs, Fussell-Vesely (F-V) and risk achievement worth (RAW) importance measures for core-damage frequency (CDF) and large early release frequency (LERF), and times available to complete each action. It is important for the NRC staff to understand which operator actions were eliminated by the screening process.

9. Attachment 6, Section 10.5.3, of the submittal dated September 10, 2003, states that of all the operator actions screened from further analysis, only three actions when assumed failed with a HEP of 1.0 would result in an increase in CDF by $\geq$ 1E-6/y or LERF by $\geq$ 1E-7/y. Please identify these operator actions and provide relevant information (HEPs, importance measures, and available times).

10. Attachment 6, Section 10.5.3, of the submittal dated September 10, 2003, does not discuss how dependent operator actions were addressed in the VYNPS PSA model. Please discuss how dependent operator actions were addressed, including details such as the process used to identify dependent operator actions and the method used to develop the joint HEP.

11. Attachment 6, Section 10.5.4, of the submittal dated September 10, 2003, states that no changes were made to the system modeling due to the EPU. The original individual plant examination (IPE) submitted by VYNPS states that the plant's turbine bypass capacity is 105% of rated steam flow and that the main condenser capacity is 110% of rated steam flow. Since the EPU will increase steam flow, it appears that the EPU plant's turbine bypass and main condenser capacities have been somewhat reduced. Please discuss how these reductions affect the EPU plant's response to an anticipated transient without scram (ATWS) event. Also describe how the reduced steam dump capacity impacts the reactor trip frequency.

12. Attachment 6, Section 10.5.4, of the submittal dated September 10, 2003, states that the change in LERF is due to the change in CDF. However, the original IPE submitted by VYNPS indicates that the results of the Level 2 PSA depend on various key operator actions. Did the screening process used to identify operator actions for adjustment consider actions specific to the Level 2 PSA? Were the Level 2 PSA results recalculated to reflect changes to operator actions specific to the Level 2 PSA ?

13. Attachment 6, Section 10.5.4, of the submittal dated September 10, 2003, does not discuss the potential impact of the increased decay heat due to the EPU on the accident progression and containment event tree (CET) quantification. Please summarize any calculations (MAAP runs) performed for the EPU plant to confirm that the existing accident progression and CET modeling did not require any modifications. Was the release binning process, which depends in part on the timing of containment sequences and, hence, the decay heat load, re-evaluated?

14. Attachment 6, Section 10.5.5, of the submittal dated September 10, 2003, indicates that a qualitative evaluation of the VYNPS fire risk profile due to the EPU was based on a review of the VYNPS fire PSA performed as part of the individual plant examination - external events (IPEEE). The information in the EPU submittal is, in fact, a quantitative analysis that adjusted the original IPEEE fire analysis results using numerical results from the internal events PSA of the EPU plant. The original IPEEE fire analysis was based on the Fire-Induced Vulnerability Evaluation (FIVE) developed by EPRI The FIVE methodology requires both qualitative and quantitative screening of fire areas and plant responses. Were the results of the screening analyses performed in the IPEEE re-examined and confirmed for the EPU plant? Were the plant modifications specific to the EPU that are listed in Section 10.5 systematically considered for their potential impact on fire risk, either through review of design documentation or plant walkdowns?

15. Attachment 6, Section 10.5.5, of the submittal dated September 10, 2003, indicates that the EPU does not impact the results of the seismic margins assessment (SMA) performed as part of the original IPEEE. Were the plant modifications specific to the EPU that are listed in Section 10.5 systematically considered for their potential impact on seismic risk, either through review of design documentation or plant walkdowns?

16. Attachment 6, Section 10.5.6, of the submittal dated September 10, 2003, states that the impact of the EPU is to increase the shutdown CDF by about 2%. How was this value estimated without performing a shutdown PSA?

**Probabilistic Safety Assessment Branch (SPSB)**

Containment and Accident Dose Assessment Section (SPSB-C)
Reviewers: Richard Lobel (Containment), Harold Walker (HVAC), Michelle Hart (Dose)

Questions to be provided later

**Reactor Systems Branch (SRXB)**

Boiling Water Reactors and Nuclear Performance Section (SRXB-A)
Reviewer: Edward Kendrick

Questions to be provided later