September 7, 2005

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

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

Dear Mr. Kansler:

By letter dated September 10, 2003, as supplemented on October 1, and October 28 (2 letters), 2003, January 31 (2 letters), March 4, May 19, July 2, July 27, July 30, August 12, August 25, September 14, September 15, September 23, September 30 (2 letters), October 5, October 7 (2 letters), December 8, and December 9, 2004, and February 24, March 10, March 24, March 31, April 5, April 22, June 2, August 1, and August 4, 2005, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., submitted a proposed license amendment to the Nuclear Regulatory Commission (NRC) for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed amendment, "Technical Specification Proposed Change No. 263, Extended Power Uprate," would allow an increase in the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWT) to 1912 MWT.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosed request for additional information (RAI). The NRC staff has determined that the RAI contains proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations*. As such, we have enclosed non-proprietary and proprietary versions of the RAI (Enclosures 1 and 2, respectively).

We request that the additional information be provided by September 9, 2005. The response timeframe was discussed with Mr. Craig Nichols of your staff on August 30, 2005. If circumstances result in the need to revise your response date, or if you have any questions, please contact me at (301) 415-1420.

Sincerely,

/RA/

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

Docket No. 50-271

Enclosures: As stated

cc w/Enclosure 1 only: See next page

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Accession Nos.:
Package: ML052430059
Letter and Enclosure 1: ML052430018 (publicly available)
Enclosure 2: ML052430038 (non-publicly available)

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED LICENSE AMENDMENT

EXTENDED POWER UPRATE

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

By letter dated September 10, 2003, as supplemented on October 1, and October 28 (2 letters), 2003, January 31 (2 letters), March 4, May 19, July 2, July 27, July 30, August 12, August 25, September 14, September 15, September 23, September 30 (2 letters), October 5, October 7 (2 letters), December 8, and December 9, 2004, and February 24, March 10, March 24, March 31, April 5, April 22, June 2, August 1, and August 8, 2005, (References 1 through 32), Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee), submitted a proposed license amendment to the Nuclear Regulatory Commission (NRC) for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed amendment, “Technical Specification Proposed Change No. 263, Extended Power Uprate,” would allow an increase in the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWT) to 1912 MWT.

The NRC staff is reviewing your Extended Power Uprate (EPU) amendment request and has determined that additional information is required to complete the review. The specific information requested is addressed below.

Electrical and Instrumentation and Controls Branch (EEIB)

Electrical Engineering Section (EEIB-A)
Reviewer: Amar Pal

6. As followup to the response to request for additional information (RAI) EEIB-A-4 in Supplement 30, Attachment 4, it appears that the direct current required to close the required breakers in order to provide an alternate alternating current (AAC) power source was not considered in the original coping analysis. Additionally, 6 amps are needed to close one breaker. However, two breakers are involved for the AAC power source. Furthermore, the spring-charging current after the breakers are closed will be much higher. Please explain why the spring-charging current is not considered in the battery capacity and voltage calculations. Are there any other loads not currently considered in the coping analysis calculation?

7. As followup to the response to RAI EEIB-A-2 in Supplement 30, Attachment 4, your response indicated that “should the SBO [station blackout] event occur during a winter snow storm that could delay VHS [Vernon Hydroelectric Station] startup, the conservatism in heat sink temperature (which assumes peak summer allowable temperature) would allow for additional coping time.” It appears from this statement that the coping time could be more than two hours during a snow storm. Please provide information regarding the worst-case coping time under any conditions and demonstrate

that the current coping analysis timeframe of two hours, and the associated conservatisms, is bounding.

In addition, the response stated “Based on their experience, which includes off hours events in which the VHS needed to be re-started, TransCanada indicated that they had restarted the unit within the required ISO-NE [ISO New England] response timeframe.” Please provide details regarding the ISO-NE response timeframe.

8. Supplement 25, Attachment 1, Table 1, provides the timeline for AAC source startup and alignment. Step 3 describes the activities associated with notifying and staffing the VHS personnel in preparation for blackstart. The time assumed for these activities is # 90 minutes. The response to RAI EEIB-A-1 in Supplement 30, Attachment 4 discusses a tabletop review of the procedures of the actions required for an SBO event. Provide additional information regarding how the tabletop review will verify this step can be accomplished in 90 minutes under worst-case conditions.

Plant Systems Branch (SPLB)

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

30. Condensate and Feedwater System
(Safety Evaluation Template Section 2.5.4.4)

In the NRC staff's RAI dated July 27, 2005, question SPLB-A-28 reads as follows:

"EPU operation will result in a substantial reduction in the available condensate and feedwater system operating margin and plant modifications must now be credited for preventing challenges to reactor safety systems that would otherwise occur upon the loss of a RFP [reactor feedwater pump] or a condensate pump [CP]. Because the plant response to loss of RFP and condensate pump events following EPU implementation is substantially different from the response at the current licensed power level, and the expected EPU response has not been confirmed by previous full-power tests or plant transients, the NRC staff requires that the power ascension test program include sufficient testing at the 100% EPU power level to confirm that the plant will respond as expected following a) the loss of a RFP, and b) the loss of a condensate pump. Please provide a complete description of the full-power testing that will be completed in this regard for the staff's review and approval, and propose a license condition that will assure that the proposed testing will be completed as described and that the results are fully satisfactory as a prerequisite for continued operation at the EPU power level."

Entergy provided a response to this question in Attachment 8 to Supplement 30. Based on the plant modifications made to the condensate and feedwater system associated with the EPU, and consistent with the guidance in Standard Review Plan (SRP) 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," Draft Revision 0, dated December 2002, the NRC staff believes that the response to RAI question SPLB-A-28 does not provide adequate justification to exclude testing of the post-EPU plant response to a loss of an RFP or a CP. Areas in the RAI response lacking sufficient justification are addressed below.

a) The last paragraph on page 5 of 12 in Attachment 8 of Supplement 30 states that:

"The operation of the feedwater and condensate systems in terms of required response to initiating events does not fundamentally change at EPU. At CLTP [current licensed thermal power] the trip of a CP requires operator action to reduce RR [reactor recirculation] flow/power level to a point supported by the remaining pumps."

Loss of feedwater is a design basis event, which is an initiating event for a reactor trip. VYNPS Updated Final Safety Analysis Report (UFSAR) Section 14.5.4.3 lists feedwater control system failures or RFP trips as being the initiating events that can lead to partial or complete loss of feedwater flow. At the CLTP, two CPs and two RFPs are capable of providing sufficient reactor feedwater flow for full power operation; there is no need to rely upon a delay circuit for tripping the RFPs sequentially on a loss of suction pressure or on an automatic RR runback to keep the reactor from tripping. This is not the case for post-EPU operation in that all three RFPs will now be running at the 100% EPU power level, and a trip of a CP could cause inadequate suction pressure and sustained loss of flow to all three RFPs such that all these pumps trip unless the low suction pressure trip timing delay feature works as designed. The runback feature apparently could credibly reduce frequency of challenges to reactor trip and associated safety systems by responding to an RFP trip; however, this feature is not currently planned to be tested at EPU conditions. Therefore, the NRC staff's concern is that, contrary to the licensee's response, the response of the feedwater system has fundamentally changed as a result of the EPU modifications, and the integrated response of the system is not currently planned to be tested to ensure that the protective features will work as designed to prevent an unnecessary challenge to critical safety functions. Thus, the licensee's response does not adequately assess the change in the integrated response of the feedwater system.

b) The fifth paragraph on page 6 of 12 in Attachment 8 to Supplement 30 states that:

"The RR runback based on a RFP or CP trip or low feedwater suction pressure does not meet any of the criteria per Attachment 2 [to SRP 14.2.1], "Transient Testing Application to Extended Power Uprates."

As discussed in Section III.B.2 of SRP 14.2.1, "...the licensee should have considered the safety impact of first-of-a-kind plant modifications, the introduction of new system dependencies or interactions, and changes in response to initiating events. The review scope can be limited to those functions important to safety associated with the anticipated operational occurrences described in Attachment 2 to this SRP, 'Transient Testing Applicable to Extended Power Uprates'." Attachment 2 (page 14.2-17) lists "Dynamic response of plant to loss of feedwater flow" as a transient test that should be considered for EPUs to demonstrate plant performance is in accordance with the design.

Following EPU implementation, the plant response to a loss of an RFP and/or CP will rely upon the automatic RR runback feature and the RFP suction pressure trip delay feature which are not needed for CLTP operation. The licensee's response indicates that the RR runback and RFP low suction pressure trip features are not functions that are important to safety. However, these features are relied upon for minimizing challenges to reactor safety systems following an RFP and/or CP trip during EPU operation. The low suction pressure time delay feature helps to ensure that EPU operation will not significantly increase the frequency of a total loss of feedwater event which is a design basis event. Therefore, contrary to the licensee's response, the NRC staff believes that the RR runback and RFP low suction pressure trip features are important to safety consistent with the guidance in Section III.B.2 of SRP 14.2.1.

Furthermore, the capability to withstand the loss of an RFP while operating at full power conditions without causing a low reactor water level scram was typically demonstrated during the initial startup test programs. For example, during the original startup test program for Browns Ferry Units 2 and 3 (BF2/3), one of the three operating RFPs was tripped and the automatic flow runback circuit acted to drop power to within the capacity of the remaining RFPs (BF2/3 UFSAR Section 13.5).

Based on the above, the licensee has not provided adequate justification to exclude testing of the post-EPU plant response to a loss of an RFP or CP as it relates to a loss of feedwater event. The licensee needs to demonstrate that plant performance will be in accordance with the design post-EPU.

c) Starting on the bottom of page 6 of 12 in Attachment 8 of Supplement 30, the RAI response addresses the acceptance criteria in SRP Section III.C.2 and states, in part, that:

"Entergy is unaware of any VYNPS or industry EPU operating experience that supports performance of this test. The operational history of VYNPS and the very limited industry experience with RFP and CP trips at power supports that there is little benefit in injecting this transient."

The RAI response attempts to justify not performing the test based on an absence of operating experience. The intent of this SRP section is for the licensee to provide operating experience (e.g., transients or actual testing) in the industry which demonstrates that the plant will respond as expected under those transient conditions for which the licensee is proposing to take exception. In order to exempt from a specified transient from EPU program based on favorable operating experience, the staff needs to determine the applicability of the operating experience to the specific plant requesting the EPU. Additionally, the staff should verify that the licensee adequately considered operating experience associated with problems, malfunctions, or other unexpected consequences from previous power uprates. The VYNPS response does not address the intent of the criteria that is described in SRP Section III.C.2. Based on the above, the licensee has not provided adequate justification to exclude testing of the post-EPU plant response to a loss of an RFP or CP as it relates to a loss of feedwater event.

Additionally, this is a test that was typically performed during original startup testing (see BF2/3 testing discussed above). However, because VYNPS only requires the flow from two RFPs during CLTP operation, the plant conditions following a trip of one of the three CPs is reasonably stable and an automatic RR runback circuit and delayed RFP low suction pressure trip circuit were not needed. For EPU operation, the RFP low suction pressure trip circuit is now necessary and must be installed to prevent a total loss of feedwater upon the loss of a CP.

In summary, EPU modifications change the condensate and feedwater pump operating conditions (i.e., number of operating pumps and flow through each pump) and the feedwater system controls (pump low suction pressure trip logic). Also, the modifications have the potential to adversely change the frequency of the total loss of feedwater event and possibly create the potential for a malfunction of a structure, system, or component important to safety (Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59 criteria for license amendments). Loss of a CP may cause a loss of all RFPs due to higher flow through the remaining CPs and inadequate RFP net positive suction head. The VYNPS modification for the RFP suction pressure trip feature addresses this concern, but it has not been tested in an integrated manner.

31. Compressed Air/Gas System

In Section 2.5 of Attachment 2 to Entergy's letter dated March 24, 2005, the licensee provided the following information regarding safety relief valves (SRVs):

"The SRVs which are used to depressurize the reactor are provided with nitrogen accumulators. Additionally, a backup N_2 [nitrogen] supply system was installed to support manual operation of the SRVs for 72 hours (Reference 5). The backup system automatically (via a check valve) provides makeup to the SRV nitrogen accumulators."

During an SBO event, how are the SRVs expected to operate? How does that expected SRV operation align with the assumptions used to size the nitrogen accumulators and the 72-hour backup nitrogen supply?

Reactor System Branch (SRXB)

Boiling Water Reactors and Nuclear Performance Section (SRXB-A)
Reviewer: Muhammad Razzaque (questions 59 - 61, 64), George Thomas (questions 62 - 63), Zena Abdullahi (questions 65 - 71)

59. The response to RAI SRXB-A-8 in Supplement 30, Attachment 9, is not clear regarding whether single loop operation of shutdown cooling (SDC) is assumed as part of the VYNPS Appendix R analysis. If single loop operation is assumed, has an evaluation been performed at the proposed EPU conditions to demonstrate that VYNPS can achieve cold shutdown, within the required time, with only a single SDC loop during an Appendix R fire event?

60. Clarify the distinction between the terms “equilibrium core,” in the response to RAI SRXB-A-10, “representative cycle core” in Section 2.2 of the VYNPS Power Uprate Safety Analysis Report (PUSAR) (i.e., Attachment 4 of the application dated September 10, 2003), and “power uprate representative equilibrium cycle core design” in the response to RAI SRXB-A-9.

61. The response to RAI SRXB-A-11 in Supplement 30, Attachment 9, states that the current licensing basis requirements for new or spent fuel storage are not being changed by the proposed EPU. However, the response does not address whether any analysis was performed regarding the affect of the proposed EPU on new and spent fuel storage. Please address whether this analysis was done and, if so, the results of the analysis. The response should address the affects of enrichments levels in new fuel, and potential increase of some elements/isotopes (such as plutonium) in spent fuels, etc.

62. The proposed changes to Technical Specification (TS) 3.4.C.3 are shown on page 8 of Attachment 1 to the application dated September 10, 2003. This TS includes a mathematical expression showing the relationship between standby liquid control (SLC) system pump flow rate, boron concentration, and boron enrichment that is required to demonstrate SLC system operability consistent with the requirements in 10 CFR 50.62(c)(4). Additional information is required to demonstrate that the proposed value of 1.29 in this mathematical expression is acceptable at EPU conditions.

63. Section 2.8.5 of the safety evaluation template in Review Standard RS-001 directs the NRC staff to evaluate the licensee’s accident and transient analyses to determine if the analyses adequately account for operation of the plant at the proposed EPU power level. Please describe the transients that are analyzed at the current licensed power level for determination of the operating limit minimum critical power ratio and discuss which transient is most limiting. In addition, please confirm that the seven transients listed in Section 9.1 of the NRC staffs safety evaluation dated March 31, 2003, for General Electric (GE) licensing topical report NEDC-33004P, “Constant Pressure Power Uprate,” will be analyzed for the first EPU core.

64. Provide the values for maximum bundle power and average power densities at VYNPS before and after the EPU.

65. Linear Heat Generation Rate (LHGR)

The NRC staff had previously asked whether any uncertainties were applied to the LHGR limit (curve) and the actual operating nodal steady state kilowatt/foot (kw/ft). The response to RAI SRXB-A-41 took credit for a reduced value in the gradient uncertainty. However, the power allocation and the pin power uncertainty values were increased to accommodate the lack of gamma scans of the current GE14 fuel designs as operated. The RAI response states that a local uncertainty of [[[redacted]]] in LHGR is assumed in the development of the LHGR, implying that the [[[redacted]]] kw/ft uncertainty addressed in the response to the staff RAI 5, associated with the NRC-approved safety limit minimum critical power ratio (SLMCPR) topical report NEDC-32694P-A, was intended for the generation of the LHGR limit. **However, it is the staff's understanding that the uncertainty analyses provided in the RAI 5 response was addressing the uncertainty to be applied to the kw/ft calculated by the core monitoring system (e.g., 3D MONICORE) as opposed to a [[[redacted]]] uncertainty assumed during the development of the LHGR curve.**

The RAI 5 to NEDC-32694P-A stated that the process computer monitors peak kw/ft and maximum average planar linear heat generation rate (MAPLHGR). The peak kw/ft and the MAPLHGR depend on the bundle axial power distribution and, consequently, are significantly more sensitive to the 3-D MONICORE replacement of the traversing incore probe (TIP)/local power range montior (LPRM) axial power distribution. The RAI asked for uncertainty analysis for the 3-D MONICORE prediction of peak kw/ft and MAPLHGR. In the response, GE provided the following uncertainty analyses, which specified the uncertainty that would be applied to the peak kw/ft calculations:

<u>Nodal Power Uncertainty:</u> The nodal power uncertainty for 3D MONICORE is a combination of: 1) the uncertainty in the four bundle power at axial node k; 2) the uncertainty in the power allocation factor at node k; 3) the LPRM update uncertainty; and 4) the uncertainty in the TIP axial power distribution at node k. [[[redacted]]] The total nodal power uncertainty is, therefore, equal to:

[[[redacted]]]

<u>Pin Power Peaking Uncertainty:</u> The pin power peaking uncertainty can be determined from the factors outlined for the R-factor uncertainty summarized in Section 3 of NEDC-32601. Specifically, the pin power peaking uncertainty is a combination of 1) the model uncertainty, 2) the manufacturing uncertainty, and 3) the channel bow uncertainty. As in Section 3 of NEDC-32601P, the model uncertainty is a combination of the pin peaking uncertainty determined from Monte Carlo comparisons (1.44%) and

an uncertainty due to flux gradients from neighboring bundles. [[ ]] All of these pin power uncertainties have been combined in NEDC-32601P as:

[[ ]]

The total LHGR uncertainty is the combination of nodal and pin power uncertainties:

[[ ]]

<u>Staff Position</u>

As shown in the NRC-approved SLMCPR methodology specified in NEDC 2694P-A, σ_{LHGR} changes with σ_{pal} and σ_{MC}. Accepting the reduction in the gradient uncertainty, a σ_{LHGR} of [[ ]] should be applied to the calculated kw/ft as discussed and specified in the NRC-approved licensing topical report. Because a [[ ]] uncertainty is assumed in the generation of the LHGR limit, this does not mean that the uncertainties due to the impact of modeling uncertainties on the operating kw/ft can be traded off with the [[ ]] uncertainty assumed in the development of the limit. The limit is developed based on the accuracy of the thermal-mechanical analytical models, methods and code systems. Therefore, any uncertainty currently applied in the development of the LHGR limit, can only be taken credit for or changed if it is demonstrated that for the current fuel designs and operating conditions additional nonconservatisims would not offset the “no cause” [[ ]] uncertainty.

The increase in the power allocation and pin power uncertainty applied to the SLMCPR does not directly lead to a proactive increase in the predicted steady state kw/ft. Therefore, potential underestimation in the nodal powers (bundle and peak pin) need to be accounted for. As evident in the RAI responses, the core-wide axial and nodal uncertainties determined through the TIP comparisons are not applied to the transient or accident analyses. The core-wide radial (e.g., bundle uncertainty σP4B) uncertainty is limited to the SLMCPR calculations. Therefore, there are no nodal or pin uncertainties that are applied to the predicted kw/ft. It is the staff’s position that a [[ ]] kw/ft uncertainty be applied to the operating kw/ft calculated in the core simulator code, because of the following reasons:

1. Since there are no measurement data to validate the bundle and pin axial power, the uncertainties in the cross-sections and the pin powers are based on the TIP four bundle readings and the MCNP/TGBLA code-to-code comparisons. The four radial bundle uncertainty $\sigma_{P4B\ nodal}$ is derived from TIP comparisons and is applied to the SLMCPR. The power allocation between the four bundles $\sigma_{PAL\ nodal}$ derived from measurement data is also applied to the SLMCPR. The predicted operating kw/ft relies on the predicted axial bundle power and the pin powers. Although the 3D MONICORE adjusts the four bundle axial power peaking to the TIP reading, the adjusted axial power peaking is based on at least four bundle TIP response.

Therefore, the power allocation in each bundle must be incorporated in the predicted kw/ft. Similarly, the uncertainty in the pin power needs to be included in the calculation of the peak kw/ft. Therefore, the calculated [[ ]] uncertainty needs to be applied to the predicted kw/ft, to account for the uncertainties in the cross-sections and the pin powers.

2. The [[ ]] power uncertainty bias, applied in the fuel rod internal pressure cited in the Alternative Approach (Supplement 30, Attachment 1), accounts for the differences between the design conditions the rod internal pressure calculations are based on and the rod internal pressures that would be obtained if actual operating history conditions were simulated. In other words, the [[ ]] uncertainty accounts for the difference between the as-designed and as-operated conditions.

3. The Alternative Approach cites an additional power uncertainty of [[ ]] power that is not specifically assigned to any cause. The Alternative Approach also states that separate experimental benchmarking information confirms that the model uncertainties remain valid. However, it is the NRC staff's understanding that, for the current fuel designs (GE14) as operated, no benchmarking of the fission gas inventory was performed. It is also the understanding that the [[ ]] "no cause" uncertainty is based on the original NRC-approval of the thermal-mechanical methodology and models. Therefore, it is not evident if a conservatism of [[ ]] would actually be available, if the operating and core design changes implemented since the initial development of the fuel thermal-mechanical models are evaluated. Neither the RAI response nor the Alternative Approach demonstrated this. The RAI response also did not discuss what uncertainties are assumed in the transient overpower kw/ft and if there is sufficient margin available.

4. The application of [[ ]] margin to the calculated kw/ft values would ensure that there are sufficient margins to the pellet exposure limits. The [[ ]] additional margin in the peak kw/ft would require a decrease in the nodal (bundle-wise) operating kw/ft, which would provide additional margin in bundle averaged accumulated exposure.

66. CASMO/TGBLA04 Code-to-Code Comparisons

In the June 30, 2005, meeting with Entergy, the NRC staff discussed with the licensee the need for code-to-code comparisons to confirm GENE's lattice physics code capability with depletion. Currently, GE uses MCNP to perform the code-to-code comparisons without coupling MCNP calculations with an independent depletion code. Therefore, the uncertainties and the biases of TGBLA are established using MCNP with isotopic concentration from TGBLA to account for depletion effects. This approach provides the inherent bias and uncertainties of the TGBLA methods and data assuming the isotopics concentrations and excluding the effects of errors in the depletion calculations. Therefore, the uncertainties are developed using TGBLA/MNCP comparisons. Considering the lack of measurement data for the current fuel design as operated, Entergy is in a position to perform lattice physics code-to-code benchmarking using CASMO4. From the July 12, 2005, telephone call, the NRC staff understood that Entergy was going to perform code-to-code lattice physics data comparisons. However, the licensee failed to provide the CASMO4/TGBLA lattice physics data code-to-code comparison. Core follow thermal limits comparisons of TGBLA/PANACEA and CASMO4/SIMULATE-3/JAFCPR 2.1 were provided. The staff finds the thermal limits comparisons useful; however, the main task at hand should have been to provide the independent code-to-code benchmarking of the standard GE TGBLA lattice physics method. Specifically, a code-to-code method would provide a means to evaluate the errors associated with the standard GE fit/extrapolation method.

a) As originally stated, provide code-to-code comparisons for some of the limiting lattices in terms of bundle powers, enrichment and gadolinium loading. Provide plots of the lattice code-to-code cross-section and pin power peaking and isotopic inventory comparisons. Provide plots comparing the same neutronic parameters as those included in MFN 04-026, Enclosure 3. Perform these comparisons on a lattice basis. Alternatively, state why CASMO/TGBLA code-to-code comparisons were not, or cannot, be provided. Note that errors in the cross-sections affect the predicted bundles powers, the nodal (bundle wise) axial power peaking and profiles and the changes in the core reactivity with change in the voids during anticipated operational occurrences (AOOs) and accident conditions. While it is difficult to reconcile differences in the cross-sections (e.g., flux ratios) between two independent depletion codes, the differences and trending are useful in evaluating the capability of the code being assessed. In particular, if the independent code predictions are supported by comparisons to measured data (bundle and pin gamma scans) based on current fuel designs and operated at the current conditions, then such comparisons are valuable as an interim process. The reason for seeking the CASMO-4/TGBLA comparisons is that MCNP is not a depletion code.

b) Provide additional information on the uncertainties applied in the CASMO4/SIMULATE-3/JAFCPR2.1 calculations. State if the Simulate-3 uncertainties are based on LPRMs or TIP-based uncertainties.

67. Shutdown Margin (SDM)

In the Alternative Approach and in the RAI responses, VYNPS SDM data was not provided as discussed in the July 12, 2005, telephone conference. As the NRC staff pointed out in the June 30, 2005 meeting, Figure 25-18, "Cold Critical Eigenvalues-All Cycles Studies," of the MFN-05-029 shows that the actual cold eigenvalue tracking of different plants show a scatter of the bias of each plant. However, the uncertainty applied to each plant is obtained by RMS averaging of bias from all plants. Thus, it seems that a bias of 0.38% $\Delta k/k$ is applied to the calculated core-wide critical keff (in-sequence cold eigenvalue) although the bias from critical (keff = 1.0) may be larger for a given plant. Also, presenting the calculated cold critical eigenvalue alone does not indicate if the critical control rod positions were predicted.

a) Provide the VYNPS cold critical eigenvalues for at least two cycles. Include the recent mid-cycle startup cold critical eigenvalue. Include tables of the predicted keff with the CR withdrawals and indicate predicted critical eigenvalue and the calculated cold critical eigenvalue corresponding to when the core became critical. Evaluate the bias in the VYNPS cold critical eigenvalue data.

b) Provide the actual calculated SDM, with the correction for the period, temperature and peak reactivity.

c) The alternative approach states that for VYNPS "the standard design SDM is 1.1% Δk/k to provide additional flexibility in cycle length and operations." Clarify this statement. Is this an additional margin included to meet the cycle energy needs or is this additional conservatism that ensures SDM for any point in the cycle?

d) The Alternative Approach did not include impact of potential underprediction in reactivity and bundle and pin powers on the SLC system cold shutdown capability. Provide an evaluation of the SLC system shutdown capability and rod withdrawal error analysis.

e) Demonstrate that the [[ ]] would not have an important impact when the [[ ]] void fraction and extrapolation to higher voids are used. Also, provide a discussion on what such an under-prediction would have on the accuracy of the local reactivity predictions and what impact, if any, it would have on the SDM, SLC system cold shutdown and rod withdrawal error calculations.

f) The RAI responses stated that the objective is for the eigenvalue trendline to remain constant and consistent from cycle to cycle for a given plant, unless significant change in core loading design results in some change in the trendline. However, the trendline is not a licensing parameter and can be adjusted according to a new trendline fitting a change in the data. The licensing parameter is the SDM. Therefore, from a licensing and safety perspective, the difference between the calculated keff for a critical reactor and the deviation from 1.0 is the most important parameter. Explain why it is not desirable for the keff bias and uncertainty to be derived on plant-specific bases. Thus, ensuring a better adjustment applied to the keff bias assumed in the SDM calculations would be based on individual plant's characteristic response and the accuracy of the neutronic methods.

68. Void Reactivity Coefficient

RAI SRXB-A-51 asked that Entergy provide an evaluation that demonstrates that the void reactivity coefficients are applicable and are developed for the range of core thermal-hydraulic conditions expected for the transient and accident conditions, including anticipated transients without scram (ATWS). The RAI response did not explicitly address the NRC staff question. The response instead discussed the conservative axial power distribution that is assumed (HBB and UB) that minimizes the scram reactivity worth. **However, the staff RAI was focused on assessing ODYN's capability to simulate the change in core reactivity with the change in voids for the current EPU fuel and core designs. In addition, the objective of the RAI is also to determine if the void reactivity coefficient bias and uncertainty derived in the original ODYN licensing topical report remains valid and applicable for the EPU core and fuel designs.**

The RAI response also referred to a sensitivity study performed during the initial ODYN licensing (NEDO-24154P-A, Volume III, page Q12) based on the Peach Bottom turbine trip transient simulation. The void coefficient was changed by [[ ]]. The sensitivity studies determined the impact changes in the void coefficient would have on the ΔCPR/ICPR response. The document concludes that a model uncertainty due to void reactivity response of [[ ]] is assumed. This sensitivity study [[ ]] It is also not clear that the void reactivity coefficients for the current fuel and core design are [[ ]]

As stated in the RAI response, it is true that ODYN AOO response is [[ ]] than TRACG. While TRACG applies a [[ ]]
Therefore, for the current EPU high energy core designs and the associated core thermal-hydraulic conditions, an uncertainty analyses is necessary in order to access the code's capability to model the changes in the core reactivity changes with changes in the void fractions.

The response to the staff's RAI 38 of the initial ODYN licensing topical report (NEDO-241154P-A, Volume 1) provides a void reactivity coefficient uncertainty analysis. The lattice k_∞ values at the three void fractions of [[ ]] The following questions relate to the appropriateness of [[ ]] used in deriving the uncertainties and biases associated with the void reactivity coefficients.

a) Provide an uncertainty analyses of the changes in the core reactivity with changes in the void fractions. Include in the uncertainty analyses how the adequacy of ODYN's predications of the reactivity coefficients can be assessed for the current EPU fuel/ core designs and operating strategy.

b) The lattice void reactivity coefficient is [[ ]] Justify the use of [[ ]] for the derivation of the uncertainties for high void conditions.

c) Provide plots showing the linear void reactivity coefficient function extended to the higher void conditions for limiting lattices in your uncertainty analysis. Include plots providing the void coefficient changes with depletion at different void conditions for the full range of instantaneous void fractions. Evaluate the changes seen in the void coefficient values with the historical void fractions for the range of the instantaneous void fractions, using limiting GE14 lattices. Based on these plots, explain the void coefficient uncertainties that would be associated with the higher void conditions for the different historical void fraction cases.

d) The response to the staff's RAI 12 of NEDE-24154P-A (page Q12-4) Volume II states that a void coefficient uncertainty of [[ ]] is applied as presented in RAI 38 (Volume I). However, the response to question 38 (page Q38-4) states that, [[ ]] Explain this statement and state if any uncertainty is applied to the void coefficient in ODYN. If so, justify why the void coefficient calculational method currently employed in ODYN, if any, is [[ ]] for the core thermal-hydraulic conditions EPU boiling water reactors (BWRs) would experience and justify the uncertainties currently used in ODYN.

e) Provide a discussion of how the changes in the void coefficient uncertainties as seen from the lattice data would affect the different transient events, instability and ATWS response.

69. Void Fraction Uncertainties

RAI SRXB-A-54 asked the following, "An EPU or a high density plant can have an exit void fraction of [[ ]] Do these void fraction predictions include the [[ ]] uncertainties in the corresponding water density calculations?"

The RAI response stated that the uncertainty in the void fraction impacts the flow and power distributions. The response states that an uncertainty is not added to the void fraction because the core follow TIP comparisons would have indicated any inaccuracies in the void fraction calculations. This RAI response did not provide sufficient justification. As discussed in response to RAI SRXB-A-36, the TIP response has many contributors and the core follow data does not provide the level of accuracy required to account for under-prediction in the nodal void fractions. In addition, the predicted void fraction is used in the offline safety analyses. The following requests address the basis for assuming no uncertainty in the void fraction calculation.

a) State if the void fraction calculations were benchmarked against measured data for all of codes that predict the void fractions and are used in the safety analyses, supporting the VYNPS EPU (e.g., PANACEA/ODYN/ISCOR/TASC). Demonstrate that the void fraction errors are insignificant or discuss the void fraction uncertainties assumed in the applicable codes. Justify why the current uncertainty is acceptable and applicable for the ranges to which it is being applied.

b) The core monitoring system was never reviewed and approved by the NRC. However, many of the RAI responses seem to qualify the impact of the higher void conditions on VYNPS by stating that the void fraction would be limited to specific value. However, no uncertainties were assumed in the predicted void fraction. If no void fraction measurement validation is available, then apply the [[ ]] uncertainty until such data can be used to demonstrate the accuracy of the prediction of the void fraction.

70. The response to RAI SRXB-A-55 did not fully answer the question. Explain why it is acceptable to exceed the void-quality correlation ranges. Provide the plot that shows the void fractions behavior at the high void conditions or quality behavior.

71. In the response to RAI SRXB-A-6, the licensee stated “the reactivity events are analyzed with the steady state tools and the results presented regarding steady-state methods in this response are directly applicable. There are some increases in power, which are significant but remain within the comparisons between the above plants for corresponding events.” This RAI response does not provide sufficient detail. The response to RAI SRXB-A-57 requested clarification to the above quoted statement. The generic event sequence was described, rather than explaining the statement in the initial RAI response. Please explain the intent of the statement in the initial submittal.

REFERENCES

1) Entergy letter (BVY 03-80) to NRC dated September 10, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Extended Power Uprate"

2) Entergy letter (BVY 03-90) to NRC dated October 1, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 1, Extended Power Uprate -Technical Review Guidance"

3) Entergy letter (BVY 03-95) to NRC dated October 28, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 2, Extended Power Uprate - Grid Impact Study"

4) Entergy letter (BVY 03-98) to NRC dated October 28, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 3, Extended Power Uprate - Updated Information"

5) Entergy letter (BVY 04-009) to NRC dated January 31, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 4, Extended Power Uprate - NRC Acceptance Review"

6) Entergy letter (BVY 04-008) to NRC dated January 31, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 5, Extended Power Uprate - Response to Request for Additional Information"

7) Entergy letter (BVY 04-025) to NRC dated March 4, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 6, Extended Power Uprate - Withholding Proprietary Information"

8) Entergy letter (BVY 04-050) to NRC dated May 19, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 7, Extended Power Uprate - Confirmatory Results"

9) Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information"

10) Entergy letter (BVY 04-071) to NRC dated July 27, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 9, Extended Power Uprate - Revised Containment Overpressure Envelope"

11) Entergy letter (BVY 04-074) to NRC dated July 30, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 10, Extended Power Uprate - Response to Request for Additional Information"

12) Entergy letter (BVY 04-081) to NRC dated August 12, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 11, Extended Power Uprate - Response to Request for Additional Information”

13) Entergy letter (BVY 04-086) to NRC dated August 25, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 12, Extended Power Uprate - Revised Grid Impact Study”

14) Entergy letter (BVY 04-097) to NRC dated September 14, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 13, Extended Power Uprate - Response to Steam Dryer Action Items”

15) Entergy letter (BVY 04-098) to NRC dated September 15, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 14, Extended Power Uprate - Response to Request for Additional Information”

16) Entergy letter (BVY 04-100) to NRC dated September 23, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 15, Extended Power Uprate - Response to Steam Dryer Action Item No. 2”

17) Entergy letter (BVY 04-101) to NRC dated September 30, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 16, Extended Power Uprate - Additional Information Related to Request for Additional Information EMEB-B-5"

18) Entergy letter (BVY 04-107) to NRC dated September 30, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 17, Extended Power Uprate - Response to Request for Additional Information related to 10 CFR 50 Appendix R Timeline”

19) Entergy letter (BVY 04-106) to NRC dated October 5, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS [emergency core cooling system] Pump Net Positive Suction Head Margin”

20) Entergy letter (BVY 04-109) to NRC dated October 7, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 19, Extended Power Uprate - Initial Plant Test Program”

21) Entergy letter (BVY 04-113) to NRC dated October 7, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 20, Extended Power Uprate - Meeting on Steam Dryer Analysis”

22) Entergy letter (BVY 04-129) to NRC dated December 9, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 21, Extended Power Uprate - Steam Dryer Power Ascension Testing”

23) Entergy letter (BVY 04-131) to NRC dated December 8, 2004, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 22, Extended Power Uprate - 10 CFR 50 Appendix R Timeline Verification”

24) Entergy letter (BVY 05-017) to NRC dated February 24, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 23, Extended Power Uprate - Response to Request for Additional Information”

25) Entergy letter (BVY 05-024) to NRC dated March 10, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 24, Extended Power Uprate - Response to Request for Additional Information”

26) Entergy letter (BVY 05-030) to NRC dated March 24, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 25, Extended Power Uprate - Station Blackout and Appendix R Analyses”

27) Entergy letter (BVY 05-034) to NRC dated March 31, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 26, Extended Power Uprate - Steam Dryer Analyses and Monitoring”

28) Entergy letter (BVY 05-038) to NRC dated April 5, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 27, Extended Power Uprate - Dryer Acoustic Load Methodology Benchmark”

29) Entergy letter (BVY 05-046) to NRC dated April 22, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 28, Extended Power Uprate - Response to Request for Additional Information”

30) Entergy letter (BVY 05-061) to NRC dated June 2, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 29, Extended Power Uprate - Computational Fluid Dynamics”

31) Entergy letter (BVY 05-072) to NRC dated August 1, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 30, Extended Power Uprate - Response to Request for Additional Information”

32) Entergy letter (BVY 05-074) to NRC dated August 4, 2005, “Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 31, Extended Power Uprate - Response to Request for Additional Information”