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September 10, 2005

Docket No. 50-271
BVY 05-083
TAC No. MC0761

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 32
Extended Power Uprate – Additional Information

References:

1) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate," BVY 03-80, September 10, 2003

2) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

3) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," July 27, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

The attachments to this letter provide supplemental information in response to requests for additional information from the NRC staff (Reference 2) and other supplemental information to update the application for a license amendment. As a result of a recent audit of certain analytical methodologies of General Electric (GE) that are used for the design and evaluation of VYNPS' fuel, the NRC staff identified the need for additional information reflected in several of the requests for additional information (RAIs) contained in Reference 2. Because of the recency of the requests, the attached is only a partial response to the Reference 2 RAIs; the remaining RAIs will be addressed in a submittal that will be made by September 16, 2005.

A001

Reference 1 discussed the plant modifications necessary to support the extended power uprate (EPU) of VYNPS and a planned two-step power increase to 120% of currently licensed thermal power (CLTP). The two step process was necessary because EPU-enabling plant modifications were scheduled to occur during two refueling outages—Spring 2004 (RFO-24) and Fall 2005 (RFO-25). Modifications completed during RFO-24 support an approximate 15% increase in reactor thermal power, and modifications planned for RFO-25 support the achievement of the full power uprate to 1912 MWt. Because the modifications necessary to support full EPU will be completed during RFO-25, VYNPS will be able to implement the ascension to 120% CLTP in one step (subject to the limitations that may be imposed as part of power ascension testing). Upon startup from RFO-25 the plant modifications necessary to achieving a full power uprate to 1912 MWt will be complete.

In addition, to update the application, it should be noted that VYNPS will complete its transition to the GE14 fuel design during the upcoming RFO-25.

Attachments 1-3 concern regulatory commitments that have either been fulfilled, or will be during future RFOs. Attachment 4 provides an updated response to RAI SRXB-A-17 that was posed in Reference 3. Attachments 5-8 provide responses to RAIs in Reference 2.

Certain Reactor Systems Branch RAIs and responses thereto in Attachment 5 contain Proprietary Information as defined by 10CFR2.390 and should be handled in accordance with the provisions of that regulation. Attachment 5 is considered to be Proprietary Information in its entirety. Attachment 6 is a non-proprietary version of Attachment 5. An affidavit provided by General Electric Company, supporting the proprietary nature of the document, is provided as Attachment 9.

There are two new regulatory commitments contained in this submittal associated with modifications to sampling probes in the condensate and feedwater systems, and future steam dryer inspections. The commitments are summarized in Attachment 10.

The following attachments are included in this submittal:

Attachment	Title
1	Steam Dryer Inspections
2	Feedwater Sample Probes
3	Motor-Operated Valve Program Commitment
4	Revised Response to RAI SRXB-A-17 Rod Withdrawal Error Transient
5	Responses to RAIs SRXB-A-59, 60, 61, 62, 63, 64, 66, 69, and 70 (Proprietary Information)
6	Responses to RAIs SRXB-A-59, 60, 61, 62, 63, 64, 66, 69, and 70 (Non-Proprietary Version)
7	Responses to RAIs EEIB-A-6 through EEIB-A-8
8	Responses to RAIs SPLB-A-30 and 31
9	GE Affidavit
10	New Regulatory Commitments

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

Entergy stands ready to support the NRC staff's review of this submittal and suggests meetings at your earliest convenience to resolve any remaining issues. If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 10, 2005.

Sincerely,

Norman L. Rademacher
Director, Nuclear Safety Assurance
Vermont Yankee Nuclear Power Station

Attachments (10)

cc: Mr. Richard B. Ennis, Project Manager
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BVY 05-083
Docket No. 50-271

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Steam Dryer Inspections

Total number of pages in Attachment 1 (excluding this cover sheet) is 2.

Steam Dryer Inspections

In Supplement 26[1] to the license amendment request, Entergy proposed certain plans for monitoring the performance of the steam dryer at extended power uprate (EPU) conditions, including an ongoing steam dryer inspection program. The intention was to formally adopt steam dryer monitoring and inspection actions as a condition to the license amendment for EPU. The commitments made in Supplement 26 were contingent upon receipt of a license amendment for EPU[2].

Because Entergy will not receive approval for EPU prior to the Fall 2005 refueling outage (RFO), the plans regarding future steam dryer inspections are being updated herein. Originally, Entergy had assumed power operation at partial uprate conditions prior to the 2005 RFO.

In Table 3 of Attachment 2 to Supplement 26, Entergy proposed conducting a visual inspection of all accessible, susceptible locations of the steam dryer during each of the next three RFOs, beginning with the 2005 RFO. Furthermore, the results of the inspections conducted would be reported to the NRC staff following startup from the respective outages. Because no power operation above originally licensed thermal power levels will occur prior to the next RFO, Entergy is modifying its steam dryer inspection plans.

Consistent with GE SIL 644, Rev. 1[3], Entergy performed a baseline visual inspection of all accessible, susceptible locations of the steam dryer during the last RFO. This action fulfills SIL recommendation A.1 and also meets the intent of SIL recommendation B.1 to:

> *Perform a baseline visual inspection of the steam dryer at the outage prior to initial operation above the OLTP or current power level.*

To clarify Entergy's intentions regarding steam dryer inspections during the upcoming refueling outage, Entergy commits to performing a visual inspection of the steam dryer modification, flaws left "as-is," and the repair made during the last RFO. This satisfies SIL 644, Rev. 1 recommendations A.1.c and A.1.d. If new indications are detected during the inspection of the modification and repair, Entergy will make a report to the NRC within 60 days following startup from the RFO.

In addition, to satisfy SIL 644, Rev. 1 recommendation B.2, Entergy will also conduct steam dryer inspections during the RFOs that occur following each of the first two full operating cycles

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[1] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, *Technical Specification Proposed Change No. 263 – Supplement No. 26, Extended Power Uprate –* Steam Dryer Analyses and Monitoring," BVY 05-034, March 31, 2005

[2] Other communications between Entergy and the NRC staff also indicate plans to conduct steam dryer inspections during the 2005 RFO (e.g., Entergy's response to RAI EMCB-A-2 that was included in Supplement 8, dated July 2, 2004)

[3] GE Nuclear Energy, Services Information Letter, SIL No. 644, Revision 1, "BWR Steam Dryer Integrity," November 9, 2004

after the final uprated power level has been achieved. Entergy now plans on conducting a visual inspection of all accessible, susceptible locations of the steam dryer during each of the three RFOs, beginning with RFO-26 (i.e., spring 2007).

Therefore, Entergy's commitment to perform visual inspections of the steam dryer is modified as stated herein.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Feedwater Sample Probes

Total number of pages in Attachment 2 (excluding this cover sheet) is 1.

Condensate and Feedwater Systems Sample Probes

In Supplement 15 to the license amendment request for EPU,[1] Entergy provided in Table 2-1 of Attachment 1 a review of industry flow induced vibration (FIV) events and their applicability to VYNPS. On page 7 of 10 (Attachment 1, Table 2-1) INPO Event # 237-031009 identifies the failure of certain sample probes in the condensate and feedwater systems. Entergy stated in Table 2-1 that VYNPS does not have sample probes in the condensate and feedwater systems.

It has been subsequently determined, however, that VYNPS has isokinetic sampling probes in the main steam, condensate, and feedwater systems. Those probes are subject to the effects of FIV, and their susceptibility to high cycle fatigue failure has been evaluated. As a result of the evaluation, Entergy will modify the four susceptible probes (identified as SP-26, 27, 30, and 31) during the upcoming refueling outage to address the failure vulnerability.

[1] Entergy letter to NRC "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," (BVY 04-100), September 23, 2004

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Motor-Operated Valve Program

Total number of pages in Attachment 3 (excluding this cover sheet) is 1.

Motor-Operated Valve Program

In Supplement 16 to the license amendment request for EPU,[1] Entergy committed to revise the Motor Operated Valve Periodic Verification Program (MOV PVP) to include periodic at-the-valve testing as a means to verify the effectiveness of the motor control center (MCC) testing methodology and to formalize the process for trending DC motor performance. This commitment has been satisfied in that the above stated actions are now complete.

[1] Entergy letter to NRC, "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," BVY 04-101, September 30, 2004

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Revised Response to RAI SRXB-A-17

Total number of pages in Attachment 4 (excludina this cover sheet) is 2.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides a revised response to the NRC Reactor Systems Branch's (SRXB) request for additional information (RAIs) in NRC's letter dated July 27, 2005.[1] Subsequent to making the response to RAI SRXB-A-17 in Entergy's letter of August 1, 2005,[2] discussions were held with the NRC staff that resulted in this revision

The individual RAI is re-stated as provided in NRC's letter of July 27, 2005.

RAI SRXB-A-17

In Supplement 4, Attachment 5, Matrix 8, page 13, note for SE Section 2.8.5.4.1, there is an explanation for uncontrolled control rod withdrawal from a subcritical or low power startup condition. In this explanatory section, this event is considered as an accident and a fuel enthalpy of 170 calories/gram is given as the acceptance criterion. However, in SRP Section 15.4.1, this event is considered as a transient, not as an accident, and hence specified acceptable fuel design limit criteria is applied. Why is this event considered as an accident rather than a transient?

Response to RAI SRXB-A-17

(The following response supersedes the response to RAI SRXB-A-17 that was provided in license amendment request, Supplement 30, Entergy's letter of August 1, 2005, BVY 05-072)

Consistent with the SRP, this event is indeed considered a transient event, not an accident.

The transient thermal limits are established such that no fuel damage is to occur during the most severe abnormal operating transient. Fuel damage is defined as perforation of the cladding that permits release of fission products. Fuel damage can occur due to two primary mechanisms: (1) severe overheating of the fuel cladding caused by inadequate cooling, and (2) fracture of the fuel cladding due to stresses which may be induced by the relative expansion of the fuel pellet inside the cladding.

To achieve severe overheating of the cladding due to inadequate cooling, it would be necessary to generate more thermal power (heat) in the fuel than can be adequately transferred through the cladding to the coolant. Transients that can cause this type of behavior, typically occur

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[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. . (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," July 27, 2005

[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 30, Extended Power Uprate – Response to Request for Additional Information," BVY 05-072, August 1, 2005

during higher power operation. Operation within the Operating Limit Maximum Critical Power Ratio (OLMCPR) protects against this.

At lower power, rapid fission gas generation and pellet expansion induced cladding stresses are a concern. In order to protect against events of this type, including the Continuous Rod Withdrawal during Startup transient, a criterion was developed that limited peak fuel enthalpy below the cladding stress failure limit.

For the Continuous Rod Withdrawal during Reactor Startup transient, NEDO-23842[3] establishes a peak fuel enthalpy licensing basis criterion of 170 cal/gm that shall not be exceeded. This criterion was adopted from NEDO 10527,[4] which states that this value is the fuel cladding failure threshold. This criterion is widely used by operating BWRs, and its use has been accepted by NRC. In fact, NUREG 1433[5] Section B3.3.1.1 states "to demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (Ref. 4) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM." The "(Ref. 4)" from this section of NUREG 1433 is NEDO-23842.

VYNPS Updated Final Safety Analysis Report[6] (UFSAR) Section 14.5.3.2, "Continuous Rod Withdrawal during Reactor Startup," states that the peak fuel enthalpies resulting from this event are less than 60 cal/gm, which is significantly less that the licensing basis limit of 170 cal/gm. As such, this is VYNPS' current licensing basis for this event, and it is not being changed for EPU. Because this event is considered a non-limiting transient, it is not required to be analyzed for EPU per NEDO-33004-A,[7] as approved by the NRC in a safety evaluation dated March 31, 2003. However, VYNPS did perform an evaluation of the Continuous Rod Withdrawal during Reactor Startup transient for EPU.

For EPU by itself, peak fuel enthalpy is not expected to increase. However, indirectly, EPU fuel and core designs may lead to higher rod worth and, therefore, higher peak fuel enthalpy at low power. It was conservatively assumed that a 20% increase in rated power would increase peak fuel enthalpy at low power by 20%, resulting in a peak fuel enthalpy for the Continuous Rod Withdrawal during Reactor Startup of 72 cal/gm, still far below the peak fuel enthalpy limit of 170 cal/gm.

[3] NEDO-23842, R.C. Stirn & J.F. Klapproth, "Continuous Control Rod Withdrawal Transient in the Startup Range," April, 18, 1978

[4] NEDO-10527, C.J. Paone, R.C. Stirn, & J.A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors," March 1972

[5] NUREG-1433, Revision 3.0, "Standard Technical Specifications General Electric Plants, BWR/4," June 2004

[6] Updated Final Safety Analysis Report (UFSAR), Vermont Yankee Nuclear Power Station, Revision 19

[7] NEDO-33004-A, Revision 4, "Licensing Topical Report, Constant Pressure Power Uprate," July 2003

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Responses to RAIs SRXB-A-59, 60, 61, 62, 63, 64, 66, 69 and 70

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Total number of pages in Attachment 6 (excluding this cover sheet) is 54.

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT
VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides responses to the NRC Reactor Systems Branch's (SRXB) individual requests for additional information (RAIs) in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. In certain instances the intent of individual RAIs may have been modified based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The individual RAIs are re-stated as provided in NRC's letter of September 7, 2005.

RAI SRXB-A-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?

Response to RAI SRXB-A-59

Single loop operation of RHR shutdown cooling (SDC) is assumed for decay heat removal as part of the VYNPS Appendix R analysis in order to achieve cold shutdown within the time required by Appendix R (i.e., 72 hours). An underlying assumption in the Appendix R analysis is that one loop of RHR is unavailable due to the postulated event.

Section 3.10.1 of the VYNPS Power Uprate Safety Analysis Report (PUSAR) discusses the SDC analysis for constant pressure power uprate (CPPU).

It should be emphasized that the design criterion cited was based on using both RHR heat exchangers, and the requirement to cool the reactor vessel from approximately 327° F (saturation temperature at 100 psig) to 125° F, which takes approximately 11 hours. This analysis is based on 85°F cooling water which provides only a 40° F ΔT thermal driving force with the reactor at 125°F.

The VYNPS Technical Specifications define cold shutdown as having a reactor coolant temperature of less than or equal to 212° F. When the reactor coolant temperature is at 212° F,

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

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there is a 127° F ΔT, which is approximately three times the thermal driving force as the two heat exchangers case. For the Appendix R scenario, the increased thermal driving force more than compensates for the assumed loss of one heat exchanger. Thus, the time required to achieve cold shutdown (i.e., 212°F) under the Appendix R scenario conditions is less than 24 hours.

Because of the much larger temperature difference between the assumed service water temperature (i.e., 85°F) and reactor coolant during hot shutdown conditions, heat exchanger performance is more effective; thus, the rate of cooldown is increased, and cold shutdown is achieved well within the 72-hour requirement assuming the operation of a single loop of RHR SDC. Thus, significant margin exits to achieve cold shutdown within 72 hours.

RAI SRXB-A-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.

Response to RAI SRXB-A-60

The three terms "equilibrium core," "representative cycle core," and "power uprate representative equilibrium cycle (PUREC) core design" are synonymous.

RAI SRXB-A-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.

Response to RAI SRXB-A-61

VYNPS has Technical Specification requirements that limit the effective multiplication factor, Keff, of the spent fuel pool (SFP) to less than or equal to 0.95 and to ensure that the infinite multiplication factor, Kinf, of any segment of fuel assembly stored in the SFP is less than 1.31 at 20°C.

Analysis has been performed that shows that ensuring the Kinf of any fuel segment is less than 1.31 will ensure that the SFP Keff remains below 0.95. For each reload, the fuel vendor, currently Global Nuclear Fuel (GNF), calculates the Kinf at 20°C for each different fuel lattice

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type to be utilized, as a function of void history and lattice exposure. These calculations address the change in elements/isotopes including plutonium. Results indicate that the peak Kinf occurs at zero void fraction due to quicker gadolinia burn out. Near the end of bundle life, Kinf is higher for bundles burned at higher void fractions than those bundles burned at lower void fractions. However, this Kinf is significantly less than the peak Kinf for bundles burned at zero void fraction. VYNPS ensures that the peak Kinf is less than 1.31 for all fuel lattice types used in the reload.

VYNPS has a Technical Specification requirement to limit the effective multiplication factor, Keff, of the new fuel storage facility to less than 0.90 when dry and 0.95 when flooded. The new fuel storage vault will not be used until a criticality analysis is completed that considers fire fighting foam entering the vault.

RAI SRXB-A-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.

Response to RAI SRXB-A-62

The equivalency equation in TS 3.4.C.3 conforms to the SLC system requirements of 10CFR50.62(c)(4) for anticipated transients without scram (ATWS). The EPU ATWS analysis also provides assurance that various VYNPS reactor and containment parameter acceptance criteria are met. The EPU analysis was performed using the following SLC system nominal values:

- flow rate of 40.5 gpm,
- boron concentration of 10.42 wt%, and
- boron-10 enrichment of 43%

When these values are combined with the mass ratio (628,300 lbs./401,247 lbs.), the result is slightly less than 1.29. To ensure that the EPU ATWS analysis remained bounding, the equivalency equation was modified to require meeting the more stringent value of 1.29 rather than the value of 1.

A review of the proposed change to TS 3.4.A.3 indicates that use of symbols for the subject four factor expression could be clarified. The combined use of an equal (=) sign and a greater than or equal (≥) sign for "Q" should be changed to a single greater than or equal (≥) sign. The two TS replacement pages provided at the end of this Attachment are a revised markup of the

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TS and a re-typed page. These pages should be substituted for those provided in the original application of September 10, 2003.

RAI SRXB-A-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.

Response to RAI SRXB-A-63

The transients that are analyzed at the current licensed power level for determination of the Vermont Yankee Nuclear Power Station (VYNPS) operating limit minimum critical power ratio (OLMCPR) are as follows:
[[

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In addition, the [[

]]

The above transient selection is consistent with *General Electric Standard Application for Reactor Fuel (GESTAR)*, NEDE-24011-P-A-14, June 2000; and the U.S. Supplement, NEDE-24011-P-A-14-US, June 2000. The above transients are analyzed for each VYNPS reload.

For VYNPS current operating cycle at Current Licensed Thermal Power, the limiting transient for determination of [[ ]]

Section 9.1 of the NRC safety evaluation dated March 31, 2003, for GE licensing topical report NEDC-33004P, "Constant Pressure Power Uprate," lists the following transients that will be re-analyzed at Extended Power Uprate [[

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The above transients listed in Section 9.1 of the NRC safety evaluation dated March 31, 2003, for GE licensing topical report NEDC-33004P, "Constant Pressure Power Uprate," will be analyzed for the first VYNPS EPU core. [[
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RAI SRXB-A-64

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

Response to RAI SRXB-A-64

The maximum allowable bundle power is determined by the VYNPS thermal limits that may vary from cycle to cycle. The values for maximum bundle power before and after EPU are 7.02 MWt and 7.37 MWt, respectively. This represents a 5% increase in maximum bundle power for a 20% increase in rated thermal power. The values for average power density before and after EPU are 48.9 kW/liter and 58.7 kW/liter, respectively.

RAI SRXB-A-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

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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.

Response to RAI SRXB-A-66

Response to Part (a)

As discussed during the NRC staff's audit of GE Methods on September 7, 2005, the two codes (CASMO-4 and TGBLA-6) use fundamentally different methodologies to calculate core parameters, including cross sections. One fundamental difference between the two codes is that each performs calculations using different neutron energy groups. Consequently, it is difficult to generate lattice cross sections that provide for meaningful comparisons. Therefore, those comparisons are not provided. However, comparisons of other parameters for five lattices designed for use in VYNPS Cycle 25, and identical to those presented in Supplement 30,[2] are provided for code comparison purposes. These lattice calculations were performed with what are understood to be identical inputs (temperatures, dimensions, etc.) within the known allowances of the methods. Because some of the comparisons include high void (90%) cases, the TGBLA-6 results are from the non-production version used to address the Pu-240

[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 30, Extended Power Uprate – Response to Request for Additional Information," BVY 05-072, August 1, 2005

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resonance treatment (to be discussed in the future response to RAI SRXB-A-67, part (e)). The range of data provided is 0, 40, 70, and 90% void fractions and the exposure range from 0 to 65 GWd/st. The lattices provided are as follows:

Lattice Number Designation	Lattice Type	Lattice Nuclear Name
6996	Dominant	P10DNAL453-16G6.0-100T-T6-6996
6997	Dominant	P10DNAL453-12G6.0-100T-T6-6997
6999	Vanished	P10DNAL448-12G6.0-100T-V-T6-6999
7007	Dominant	P10DNAL413-14G6.0-100T-T6-7007
7009	Vanished	P10DNAL403-14G6.0-100T-V-T6-7009

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For each lattice analyzed there are plots showing lattice K-infinity, local peaking and Pu-239 and 240 atom densities as calculated by the two codes. In addition, per audit request, Pu-241 atom density comparisons are also included. A list of these figures is provided below.

As shown in these figures, K-infinity performance is generally as expected with slight differences at low exposure due to gadolinium (Gd) burnout modeling differences. After Gd burnout, agreement between the two methods is good over the range of exposures with the exception of the 90% void cases which will be further discussed in the future response to RAI SRXB-A-67, part (e). While the lattice K-infinity differences are larger at this higher void, this lattice reactivity difference would have little core-wide effect due to both the small fraction of the core at those conditions and the limited power produced in those regions. This is demonstrated by the good

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comparisons of predicted (PANAC11 and SIMULATE) and measured axial powers in the core follow data provided previously.

For the lower void cases, local peaking, which is a comparison of the peak pin in the lattice, agrees well over the lower exposure range where lattices are generally limiting (high power). The two methods tend to deviate at higher exposure, non-limiting conditions. As in the case of K-infinity, the high void (90%) cases exhibit peaking differences earlier in exposure but, due to the little power produced by high void nodes, these differences are not considered significant to safety.

Isotopic comparisons between the two methods agree well and are considered to be within the ranges seen in similar methods comparisons (ref. ORNL-6901[3]). As is the case for the other parameter comparisons, the 90% void cases exhibit the greatest differences and, due to the little power and exposure occurring under those conditions, the differences are not considered significant to safety.

Additional Audit Question Responses:

Based upon the NRC staff's questions during the September 7, 2005, audit of GE Methods, the following additional information is provided:

Figure SRXB-A-66.6-1	Lattice 6696 K-infinity Fit Comparisons
Figure SRXB-A-66.6-2	Lattice 6697 K-infinity Fit Comparisons
Figure SRXB-A-66.6-3	Lattice 6999 K-infinity Fit Comparisons
Figure SRXB-A-66.6-4	Lattice 7007 K-infinity Fit Comparisons
Figure SRXB-A-66.6-5	Lattice 7009 K-infinity Fit Comparisons

These figures demonstrate the agreement between K-infinity as a function of exposure when calculated at 90% void and when extrapolated to 90% from a fit of 0%, 40%, and 70% cases. Both CASMO-4 and TGBLA-6 results are shown. These data indicate that the two methods are self consistent, i.e., data fitting and extrapolation in void is a reasonably accurate substitution for specific depletion calculations.

Figure SRXB-A-66.7-1	Lattice 7009 Void Coefficient Comparisons

This figure depicts the results of lattice void coefficient calculations (CASMO-4 and TGBLA-6) for a lattice located near the core exit which is a region of higher void. As noted in discussion with the NRC staff, the standard practice by GNF is to perform all instantaneous void cases from a 40% void history case. Therefore, CASMO-4 and TGBLA-6 comparisons of that practice

[3] ORNL6901, "OECD/NEA Burnup Credit Calculational Criticality Benchmark Phase I-B Results", June 1996

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are shown and indicate very good agreement between the two methods. In addition, CASMO-4 and TGBLA-6 results for equivalent calculations from the 70% void history case are shown for information. While the results show a difference in the void coefficient obtained from the two methods, the impact of those differences has been discussed with the staff and should not be taken out of the context of ultimate application; i.e., reactivity feedback is a function of both void coefficient, change in void, and local power (flux) such that most of the reactivity void feedback in transients occurs in lower void initial condition regions (nearer 40%).

Figure SRXB-A-66.8-1 RMS of Lattice 7009 Pin Power Differences

This figure depicts the results of a statistical evaluation of the differences in relative pin powers calculated by CASMO-4 and TGBLA-6 for a representative lattice for the 0, 40, 70, and 90% void history depletions. Due to differences in depletion steps (metric tons and short tons), only approximate exposure comparisons can be made and the data reflect the limited number of points. However, the general trends are evident and examination of the underlying data indicates that lower powered pins drive the differences while higher powered pins generally agree well. The peak or leading pin comparisons are most relevant in assessing fuel performance and safety and those are provided elsewhere in this response.

Tabulated data (EXCEL spreadsheets) of all this information will be transmitted under separate cover in a future submittal.

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Figure SRXB-A-66.1-1
Lattice 6996 K-infinity Comparison

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Figure SRXB-A-66.1-2
Lattice 6996 Local Peaking Comparison

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Figure SRXB-A-66.1-3
Lattice 6996 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.1-4
Lattice 6996 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.1-5
Lattice 6996 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.2-1
Lattice 6997 K-infinity Comparison

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Figure SRXB-A-66.2-2
Lattice 6997 Local Peaking Comparison

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Figure SRXB-A-66.2-3
Lattice 6997 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.2-4
Lattice 6997 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.2-5
Lattice 6997 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.3-1
Lattice 6999 K-infinity Comparison

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Figure SRXB-A-66.3-2
Lattice 6999 Local Peaking Comparison

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Figure SRXB-A-66.3-3
Lattice 6999 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.3-4
Lattice 6999 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.3-5
Lattice 6999 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.4-1
Lattice 7007 K-infinity Comparison

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Figure SRXB-A-66.4-2
Lattice 7007 Local Peaking Comparison

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Figure SRXB-A-66.4-3
Lattice 7007 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.4-4
Lattice 7007 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.4-5
Lattice 7007 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.5-1
Lattice 7009 K-infinity Comparison

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Figure SRXB-A-66.5-2
Lattice 7009 Local Peaking Comparison

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Figure SRXB-A-66.5-3
Lattice 7009 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.5-4
Lattice 7009 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.5-5
Lattice 7009 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.6-1
Lattice 6696 K-infinity Fit Comparisons

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Figure SRXB-A-66.6-2
Lattice 6697 K-infinity Fit Comparisons

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Figure SRXB-A-66.6-3
Lattice 6999 K-infinity Fit Comparisons

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Figure SRXB-A-66.6-4
Lattice 7007 K-infinity Fit Comparisons

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Figure SRXB-A-66.6-5
Lattice 7009 K-infinity Fit Comparisons

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Figure SRXB-A-66.7-1
Lattice 7009 Void Coefficient Comparisons

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Figure SRXB-A-66.8-1
RMS of Lattice 7009 Pin Power Differences

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Response to Part (b)

The CASMO-4/SIMULATE-3 data presented in the alternate approach response to RAI SRXB-A-6 included MFLCPR values calculated using Entergy's own in-house code, JAFCPR2.1. These data were intended to demonstrate the type of agreement seen between GNF and the independent methods used by Entergy. JAFCPR2.1 is a utility code that accurately reproduces the GEXL correlation results and, in the case shown, uses power distribution and core parameters fed to it from SIMULATE-3. Since this information is used for independent monitoring and verification, it is applied in a best estimate manner, without applying any uncertainties.

RAI SRXB-A-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.

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Response to RAI SRXB-A-69

Response to Part (a)

The GE design correlation (Ref. 69-1) for void fraction was derived in the seventies as an extension of the drift-flux model (Ref 69-2), based on void fraction measurements in simple geometries as well as full scale bundle data covering a wide range of conditions (See Table SRXB-A-69-1).

The measurement uncertainty in the multi-rod data is [[ ]]. The void correlation fits these data with an average error of [[ ]] and a standard deviation of [[ ]]. No trend is observed with bundle size or geometry (See Table SRXB-A-69-2). In addition to the multi-rod data, the void correlation has been qualified to simple geometry data covering a wide range of conditions (Ref. 69-1).

The void correlation is correlated as a function of Reynolds number, quality and fluid properties. Since the Reynolds number is a function of mass flux, hydraulic diameter and fluid properties, and the fluid properties are a function of pressure, the void correlation can also be correlated as a function of hydraulic diameter, mass flux, quality and pressure. The range in hydraulic diameters in the data is [[ ]], which is much larger than the range of hydraulic diameters in the fuel designs. The hydraulic diameter in recent GE fuel products ranges from [[ ]] for 8X8 fuel to [[ ]] in the fully rodded region of 10X10 fuel. In the region above the part length rods, the hydraulic diameters range from [[ ]] for 10X10 fuel to [[ ]] for 9X9 fuel. The pressure range covers atmospheric pressure to twice normal operating pressure for a BWR. The mass flux in a BWR ranges from approximately 400 kg/m^2-sec at natural circulation to approximately 1350 kg/m^2-sec at rated core flow, and it is seen that the mass flux range in the data far exceeds this range. The void fraction range in the data is from [[ ]], while a typical exit void fraction in BWR fuel ranges from [[ ]], for the average bundle, to approximately [[ ]] for a high power 10X10 fuel bundle such as GE14 under EPU conditions. In summary, the database for the void correlation covers all fuel products including 10X10 fuel and all operating ranges including EPU conditions.

The GE void fraction correlation is described in detail in the approved Reference 69-3. The qualification documented in the approved Reference 69-4, where the void correlation was compared to [[ ]] data points from the most representative full-scale bundles, yielded a standard deviation of [[ ]] in the void fraction, while the qualification against the wider set of [[ ]] data points as documented in References 69-1, 69-5 and the approved reference 69-7 yielded a standard deviation or [[ ]] in the void fraction (See Table SRXB-A-69-2).

The part length rod (PLR) is the major new feature in current fuel products. The impact of PLRs has been investigated for a 4X4 bundle for a pressure of 1 MPa and more recently for an 8X8 bundle at rated BWR pressure of 7.2 MPa (Ref. 69-7). A small increase, approximately [[ ]], was observed in void fraction downstream of the PLRs compared to the case with no PLR (See Figure SRXB-A-69-1) for the low-pressure 4X4 data. The recent more representative 8X8 data taken at normal operating pressure shows a small increase, on the order of [[

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The void correlation has been implemented into the GE design codes such as PANACEA/ODYN/ISCOR/TASC and the correct implementation of the void correlation has been demonstrated by functional testing. Therefore, the qualification of the void correlation applies for all design codes except TRACG. TRACG (Ref. 69-6) has been separately compared to a set of the same data discussed above and yielded a standard deviation of [[ ]] in the void fraction.

Finally, comparisons have been made to pressure drop data taken in the ATLAS test facility using full-scale test assemblies for all fuel products including the current 10X10 GE14 fuel. This testing covers a wide range of conditions including EPU conditions. For GE14 the bundle pressure drop was predicted with a mean error of [[ ]] and a standard deviation of [[ ]]. Since the pressure drop cannot be matched unless the void fraction is accurately predicted, these tests serve as an independent confirmation of the void correlation.

In the current licensing methodology with ODYN/TASC the modeling uncertainty is derived from the comparisons to the Peach Bottom 2 turbine trip tests (Ref. 69-4). Reference 69-4 also contained an alternate analysis where the void fraction was perturbed and the impact on the OLMCPR determined. In this alternate analysis the void fraction was perturbed by [[ ]], which bounds the uncertainty in the void correlation at the 95% confidence level. This comparison demonstrated that the uncertainty in the void correlation is covered in the current design process. This process has been repeated with the introduction of new fuel types such as 10X10 fuel. A similar approach is used for TRACG (Ref. 69-6) where the impacts of all model uncertainties including the uncertainty in the void fraction are combined in a statistical process to determine the OLMCPR at the 95% confidence level.

<u>Response to Part (b)</u>

The monitoring system is based on a best estimate calculation with PANACEA and is used to monitor that the design limits, such as the OLMCPR, are not exceeded. These design limits are determined, as discussed above, considering the model uncertainties, which include the void fraction uncertainty, e.g., the power distribution uncertainties include the effect of the void fraction uncertainty. In the application methodology these model uncertainties are explicitly considered such that bounding values for the design limits, such as the OLMCPR, are determined. In other words, an adder to cover the void fraction uncertainty is already included in the OLMCPR. Therefore, including an uncertainty in the monitoring system to account for the void fraction uncertainty would be equivalent to accounting for this uncertainty more than once and would be inappropriate. In summary, the core monitoring system is based on best estimate methods, where no uncertainties are considered, and the impacts of the uncertainties, such as void fraction uncertainty, are considered in the thermal limits to which the bundles are monitored. GE's 3D-MONICORE core monitoring system and the process by which the uncertainties are included in the limits were reviewed and approved by NRC as documented in References 69-7 and 69-8.

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References:

69-1 J. A. Findlay and G. E. Dix, BWR Void Fraction and Data, NEDE-21565, January 1977. General Electric Proprietary Information.

69-2 N. Zuber and J. A. Findlay, Average Volumetric Concentration in Two-Phase Flow Systems, ASME J. Heat Transfer, November 1965.

69-3 TASC-03A, A Computer Program for Transient Analysis of a Single Channel, NEDC-32084P-A, Revision 2, July 2002.

69-4 Letter, J. S. Charnley (GE) to H. N. Berkow (NRC), Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A, MFN-003-086, January 16, 1986.

69-5 Letter, G. Stramback (GE) to NRC, Completion of Responses to MELLLA Plus AOO RAIs (TAC No. MB6157), MFN 04-026, March 4, 2004.

69-6 TRACG Application for Anticipated Operational Occurrences (AOO) transient Analyses, NEDE-32906P-A, Revision 1, April 2003.

69-7 Methodology and Uncertainties for Safety Limit MCPR Evaluations, NEDC-32601P-A, August 1999.

69-8 Power Distribution Uncertainties for Safety Limit MCPR Evaluation, NEDC-32694P-A, August 1999.

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Table SRXB-A-69-1
Void Fraction Correlation Database

Data Source	Geometry	Hydraulic Diameter (m)	Pressure (MPa)	Mass Flux (kg/m^2-sec)	Inlet subcooling (K)	Exit quality (Max.)
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<u>Table SRXB-A-69-2</u>
Comparison Between Void Correlation and Database
(Taken from References 69-5 and 69-7)

Data Source	Data Points (N)	Average Error $\overline{Da} = \overline{a_m - a_c}$	Standard Deviation S_{Da}
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Figure SRXB-A-69-1
4X4 Void fraction Data – Sensitivity to PLR

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Figure SRXB-A-69-2
8X8 Void fraction Data – Sensitivity to PLR for Low Flow

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Figure SRXB-A-69-3
8X8 Void fraction Data – Sensitivity to PLR for High Flow

RAI SRXB-A-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.

Response to RAI SRXB-A-70

As explained in the response to RAI SRXB-A-69, part (a), the void correlation is based on void fraction data up to [[ ]], which covers the void fraction range expected for normal steady state operation and the abnormal operational occurrences that set the operating limit minimum critical power ratio (OLMCPR). A void fraction of [[ ]] is actually relatively high and typical of the conditions where boiling transition will occur in a BWR fuel bundle. Also, since the OLMCPR is determined such that boiling transition will not occur, it is highly unlikely that a void fraction of [[ ]] will be exceeded (e.g., momentarily during a transient) by any significant amount.

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For illustrative purposes, consider a one-dimensional, steady state energy balance for a BWR fuel channel. It can be shown that

$$X(z)=\frac{h_{in}-h_f}{h_{fg}}+\frac{1}{\dot{m}h_{fg}}\int_0^z \dot{q}'(\xi)d\xi, \qquad (70\text{-}1)$$

where the definition of flow quality is given by

$$X=\frac{\dot{m}_g}{\dot{m}_f+\dot{m}_g} \qquad (70\text{-}2)$$

The flow quality given by Equation 70-1 is a function of pressure (fluid properties), inlet flow rate and subcooling, and the heat addition rate. For the case of "z" equal to the exit elevation, the integral term essentially represents the channel power.

Figure SRXB-A-70-1 shows a typical plot of the void-quality relationship for a flow typical of a high power/flow ratio fuel bundle. This Figure shows the void-quality relationship for the entire range from zero to one. It should, however, be recognized that a BWR fuel bundle is designed and operated such that boiling transition will not occur during steady-state or abnormal operational occurrences, and, therefore, high void fractions, i.e., higher than [[ ]], will not occur. It would require a bundle power of approximately [[ ]] for a bundle at rated flow to reach a void fraction of [[ ]], while in reality a high power fuel bundle operates at approximately [[ ]]. A high void fraction of 1.0 is only possible for a severe accident scenario such as a loss of coolant accident. It is seen that the void-quality relationship is very flat in the high quality range and even a substantial increase in quality (substantial increase in power) would have negligible impact on the void fraction (exit void fraction). Therefore, even if the [[ ]] upper range of the void correlation were to be exceeded, no significant error will be introduced relative to the uncertainty in the void correlation, which is already included in the licensing methodology.

Another point can be inferred from Equation 70-1, together with Figure SRXB-A-70-1. The highest void fraction is at the top of the fuel bundle and is a result of the total integrated power in the bundle. The highest nodal power, however, is located well below the top of the bundle. Therefore, the nodes with the highest power will have a lesser void fraction than the maximum bundle void fraction. Similarly, in a transient event, the quality response in the fuel bundle is given by the mass and energy balance. It is evident from Figure SRXB-A-70-1 that the void response and the corresponding void reactivity feedback from a given quality response is much less at high void fractions that at low void fractions.

In summary, the GE void correlation is based on test data and covers a broad range of conditions (See the response to RAI SRXB-A-69). The correlation supports the full range of conditions expected during BWR operation, even at up-rated conditions. The correlation uncertainty is well defined, relatively small, and appropriately accounted for in the SLMCPR. It is not necessary to incorporate any additional penalties. Extrapolation beyond the test database ([[ ]] voids) is considered unusual and rare; and if required for a particular situation, the need to extrapolate would not be expected to introduce any appreciable error.

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Figure SRXB-A-70-1
Typical Void-Quality Relation at High Power/Flow Ratio

3.4 LIMITING CONDITIONS FOR OPERATION

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

3. The combination of Standby Liquid Control System pump flow rate, boron concentration, and boron enrichment shall satisfy the following relationship for the Standby Liquid Control System to be considered operable:

$$\frac{Q}{86} \times \frac{M251}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq \cancel{1}$$

1.29

where:

C = the concentration of sodium pentaborate solution (weight percent) in the Standby Liquid Control System tank

E = the boron-10 enrichment (atom percent) of the sodium pentaborate solution

Q $\cancel{=}$ 35 gpm

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≥

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

3. The boron-10 enrichment of the borated solution required by Specification 3.4.C.3 shall be tested and verified once per operating cycle.

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E = the boron-10 enrichment (atom percent) of the sodium pentaborate solution

$Q \geq$ 35 gpm

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

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Attachment 7

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Responses to RAIs EEIB-A-6 through EEIB-A-8

Total number of pages in Attachment 7
(excluding this cover sheet) is 4.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides responses to the NRC Electrical and Instrumentation and Controls Branch's (EEIB) individual requests for additional information (RAIs) in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. In certain instances the intent of individual RAIs may have been modified based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The individual RAIs are re-stated as provided in NRC's letter of September 7, 2005.

RAI EEIB-A-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?

Response to RAI EEIB-A-6

With respect to the battery requirements during the two-hour coping period associated with the station blackout event, two 4160 volt breakers are involved in aligning power from the AAC power source, and the spring-charging motor current should also be included. An evaluation was performed using a 20 amp load applied for a full minute at the end of the two-hour duty cycle, instead of the original 6 amps. The breaker closing current for each 4160 volt breaker is 6.0 amps. The breaker spring charging motor draws 10 amps, but this draw is not concurrent with the closing current. Therefore, the additional 20 amp load is conservative. The evaluation confirms that the additional load has no effect on end voltage and does not change the required battery capacity.

All other battery loads that occur during the two-hour coping period are currently considered in the coping analysis calculation.

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

RAI EEIB-A-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.

Response to RAI EEIB-A-7

The coping study assumes worst-case conditions corresponding to the design basis river water temperature of 85°F. These conservatisms are bounding as they result in a minimum coping time of two hours. The statement made in the response to RAI EEIB-A-2 in Supplement 30, Attachment 4, was an engineering judgment that a lower (e.g., winter) river water temperature would enable the plant to cope for a duration longer than two hours as suppression pool temperature is the limiting constraint. However, the assertion that this capability exists in no way implies that a coping time in excess of two hours will ever be required. The coping time of two hours is based on worst-case conditions and is bounding. The two hour coping time is adequate for all times of the year, as well as all postulated weather conditions. The VYNPS SBO coping analysis report, which is applicable for EPU conditions, was provided by Entergy letter of March 24, 2005.[2]

An integral portion of the ISO-NE system restoration procedure is the requirement that generating units having black start capability strive to achieve the fastest start time possible. ISO-NE black start units, such as the VHS units, are expected to be manned and prepared to commence generation within ninety (90) minutes of receiving instructions to initiate black start operations.[3] In addition, ISO-NE procedure OP-6[4] requires that during system restoration a high priority must be given to the restoration of off-site AC power sources to nuclear generators. It is stated in procedure OP-6: "[T]he most critical power requirement after a [system] blackout is the assurance of reliable shutdowns of nuclear generators. . . . The expeditious restoration of alternative off-site AC power sources to nuclear units is imperative to promote the continued

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[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 25, Extended Power Uprate – Station Blackout and Appendix R Analyses," BVY 05-030, March 24, 2005

[3] ISO New England Operating Procedure No. 11, "Black Start Capability Testing Requirements," effective date: May 6, 2005

[4] ISO New England Operation Procedure No. 6, "System Restoration," effective date: May 6, 2005

reliability of shutdown operations." Based on the designation of the TransCanada VHS units as black start units by ISO-NE, the procedural requirements for achieving black start, and the operating history of the VHS units, there is reasonable assurance that a VHS unit will be available within the SBO coping timeframe.

RAI EEIB-A-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.

Response to RAI EEIB-A-8

Step 3 in Table 1 of Attachment 1 to Supplement 25 provided a realistically conservative estimate of the time required to staff and prepare the VHS for black start under worst-case SBO conditions. The periodic tabletop review of this step in the power restoration sequence will provide added assurance that the VHS can be staffed and prepared to commence generation within 90 minutes of notification as specified in ISO-NE procedure OP-11 (see the response to RAI EEIB-A-7).

The ISO-NE system restoration exercise tabletop review, or a separate TransCanada/Entergy discussion of this 90-minute assumption, will include discussions with the VHS operator to confirm that the assumptions and completion times of restoration activities continue to remain valid. The activities and support elements to be reviewed include confirming that:

1. VHS black start restoration procedures support the 90-minute objective and are consistent with interfacing procedures of other participants involved in restoring AC power to VYNPS during an SBO event.
2. The assumption that the 90-minute timeframe includes off-hours response under adverse weather conditions (e.g., snow storms).
3. VHS units with black start capability have actually been black start tested in the past year and are in a condition to be black started.
4. Key operating aids used to support black start, such as telephone and radio communications, have been tested in the past year.
5. VHS on-call personnel are sufficient in number and proximity to VHS to support timeline assumptions.
6. VHS on-call personnel are subject to fitness-for-duty requirements
7. VHS on-call personnel are qualified for black start operations.
8. Future plans (if any) to modify procedures, staffing requirements or the black start units will continue to support the 90-minute objective.

Suggestions will be made when appropriate to increase time margins where situations warrant. The tabletop reviews will be interactive discussions intended to verify that the 90-minute objective can be met with reasonable assurance.

Attachment 8

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Response to RAIs SPLB-A-30 and SPLB-A-31

Total number of pages in Attachment 8
(excluding this cover sheet) is 7.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides responses to the NRC Plant Systems Branch's (SPLB) individual requests for additional information (RAIs) in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. In certain instances the intent of individual RAIs may have been modified based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The individual RAIs are re-stated as provided in NRC's letter of September 7, 2005.

RAI SPLB-A-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.

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

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.

Response to RAI SPLB-A-30

The implementation of EPU at VYNPS requires the use of installed excess capacity in the Condensate and Feedwater systems (these are not reactor safety systems). In addition, for economic reasons Entergy has installed a modification (RR Runback) and Low Suction Pressure setpoint change to maintain unit operation in the rare event of a trip of a condensate pump (CP) or reactor feedwater pump (RFP). This modification does not affect the probability of a total loss of feedwater event. In addition a change to the Low Suction Pressure Trip logic (sequential time delays) provides additional system protection/recovery time in a loss of suction pressure event; this is an operational enhancement not related to three RFP operation. This modification reduces the likelihood that the Lo-Lo trip of all three RFPs will occur under degraded pressure conditions. Significant testing was performed on the setpoints and logic of these modifications to ensure that the installation would result in the intended response to the anticipated input. Dynamic testing was not performed, nor was it included in EPU power ascension testing as it would only serve to (1) demonstrate that the intended actions occur and the plant remains on-line at reduced power level, or (2) initiate a plant trip on reactor level (low or high) which would be the response if the mod were not installed in the first place. Such a trip is presently analyzed both as a deterministic transient and in the VYNPS probabilistic safety assessment (PSA) for both current licensed thermal power (CLTP) and at full licensed extended power uprate (EPU) power.

The following discussion addresses each of the three points contained in the NRC staff's RAI, respectively:

a) The RAI refers to a **total loss of feedwater** resulting from a trip of a CP at EPU conditions without credit for the low suction pressure trip timing delay circuit. Entergy has performed steady state hydraulic modeling for various operational events (loss of RFP, loss of CP, etc.) with the following results:

1. A loss of a RFP will result in a RR runback (trip avoidance), and with two RFPs and three CPs a minimum final RFP suction pressure of 272.6 psig.

2. A loss of a CP will result in a RR runback (trip avoidance), and with two CPs and three RFPs a minimum final RFP suction pressure of 124.4 psig.

3. The RFP low suction pressure trip timing delay setpoint is 98psig +/- 2psig with time delays associated with each pump. The Lo-Lo suction pressure trip setpoint for all three RFPs (which provides ultimate Net Positive Suction Head (NPSH) protection) is 90psig +/- 2psig.

4. For each of the events noted above there is significant margin to both of the RFP low suction pressure trip setpoints and, although there is a reduction in total feedwater flow, there is no 'total' loss of feedwater during these events. Therefore, there is no increase in the probability of a total loss of feedwater at EPU conditions

The VYNPS PSA performed for EPU indicates that the impact of use of three RFPs at EPU is modeled as a delta to the turbine trip (TT) probability. For VYNPS the TT probability at CLTP is 0.55/year. The operation of three feedwater pumps at EPU is modeled as an impact on the TT frequency and is calculated to increase the TT to 0.57/year. Therefore, the use of three RFP at EPU results in less than a 4% increase of TT frequency. Changes in TT frequency are also evaluated for impact on Core Damage Frequency (CDF). The increase in CDF for the complete EPU evaluation including all modification, operator actions, and procedure changes is shown below:

CLTP CDF	7.77 E-06
EPU CDF	8.10 E-06
Delta CDF	3.30 E-07

A sensitivity case was run (Sensitivity #4) to determine the impact of increase TT frequency on CDF. In this case the TT frequency was doubled from 0.55/year to 1.10/year (Note: EPU TT frequency for three RFP operation is 0.57/year). The results of this case are summarized as follows:

Base EPU CDF	8.10 E-06
Case #4 CDF	8.32 E-06
Delta CDF	2.20 E-07

These results indicate that the increase in turbine trip frequency from the operation of three RFP at EPU is very small and any resultant impact on CDF is even smaller (less than 1% CDF increase for a TT increase from 0.55/year to 0.57/year). Consequently, the change in the integrated response of the feedwater system as a result of the EPU modifications has negligible safety significance.

b) The RAI indicates that VYNPS will rely upon the automatic RR runback feature and RFP low suction pressure trip delay feature at EPU conditions. The installation of the runback is a plant reliability modification to avoid undesirable plant trips where possible due to single component vulnerabilities. Failure of this circuitry would have the same results as not having installed the modifications and, as noted in response to Part (a) above, are included in the PSA for TT frequency and impact on CDF. This result would be a reduction in feedwater flow and a plant trip. As noted above the RFP low suction pressure trip delay feature is not relied upon for CP or RFP trip events and only provides enhanced RFP NPSH protection.

c) As noted in the response to Part (a) above, the trip of a CP or RFP at EPU will not result in a total loss of feedwater flow. The RAI provides reference to successful type testing of the RR runback circuitry (Browns Ferry Unit 2 (BF2) and Browns Ferry Unit 3 (BF3) initial startup test programs) demonstrating the viability of this modification. The testing at BF2 and BF3 validated the RR runback modification as a reactor trip avoidance measure. The reduction of the RFP low suction pressure trip setpoint was tested upon implementation and no further

testing is required. Since the RFP low suction pressure trip logic delay is a backup function to provide enhanced RFP NPSH protection and is not relied upon to maintain feedwater system operation after a pump trip (with or without successful RR Runback operation) the logic testing performed for this modification is deemed adequate and dynamic testing is not warranted.

Based upon the discussion above it is concluded that:

- A trip of a condensate pump or reactor feedwater pump will not result in a total loss of feedwater.
- The initiation of a Recirc Runback upon a CP or RFP trip is only intended to maintain plant operation and is not relied upon to prevent the total loss of feedwater flow. These are not reactor safety systems and EPU does not substantially change their modes of operation.
- There is minimal impact on TT frequency or CDF.
- The reduction of the RFP low suction pressure trip setpoint is based upon system and component capabilities and requires no additional testing.
- The RFP low suction pressure trip logic delay circuitry is a backup system not relied upon for protection of (non-safety system) feedwater capability upon a CP or RFP trip.
- The testing performed by Entergy for these modifications is adequate for their intended function and no additional testing is required.

RAI SPLB-A-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?

Response to RAI SPLB-A-31

During an SBO event the control room operators will use the SRVs to remove decay heat and depressurize the reactor so that shutdown cooling can be initiated when AC power is restored

following an assumed two-hour coping duration. SRV operation is no longer needed when RHR shutdown cooling begins. The reactor is then cooled to cold shutdown conditions in less than 24 hours.

The nitrogen accumulators, and the 72-hour backup nitrogen supply, provide SRV operation capability well in excess of that needed for SRV operation during an SBO event. As stated in UFSAR section 4.4.5, the original SRV accumulators are sized for approximately five operations with atmospheric pressure in the containment. The accumulators were supplemented by the addition of a backup nitrogen supply system consisting of compressed gas cylinders which provide enough capacity for hundreds of SRV operations. The backup system significantly increased the number of times that the SRVs could be manually operated. The number of required SRV operations during an SBO event is easily bounded by the 72 hour SRV nitrogen supply sizing criteria which results in significant nitrogen supply margin.

BVY 05-083
Docket No. 50-271

Attachment 9

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

GE Affidavit

Total number of pages in Attachment 9 (excluding this cover sheet) is 3.

General Electric Company

AFFIDAVIT

I, Louis Quintana, state as follows:

(1) I am Manager, Licensing, General Electric Company ("GE"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 2 of GE letter, GE-VYNPS-AEP-401, Response to NRC RAIs SRXB-60, 61, 63, 64, 66, 69, and 70, dated September 9, 2005. The proprietary information in Enclosure 2, Response to NRC RAIs SRXB-60, 61, 63, 64, 66, 69, and 70, is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation[3] refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions from analyses supporting the extended power uprate of the Vermont Yankee Nuclear Power Station utilizing analytical models and methods, including computer codes, and methods of applying these for safety analyses which GE has developed. The development of these models and computer codes and methods was achieved at a significant cost to GE, on the order of several million dollars.

The development of the analytical methods and evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 9th of September 2005.

Louis M Quintana

Louis Quintana
General Electric Company

Attachment 10

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

New Regulatory Commitments

Total number of pages in Attachment 10
(excluding this cover sheet) is 1.

Licensee Identified Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

Table 10-1

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE[1] (If Required)
	One-Time Action	Continuing Compliance	
Visual inspection of steam dryer	X		RFO-26, RFO-27, and RFO-28
Modification of the four susceptible isokinetic sample probes in the condensate and feedwater systems	X		Next scheduled refueling outage (RFO-25)

[1] Refueling outages (RFOs) 25, 26, 27 and 28 are expected to occur during the fall 2005, spring 2007, fall 2008, and spring 2010, respectively.