Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500
Tel 802 257 5271

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Docket No. 50-271
BVY 05-107
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. 43
Extended Power Uprate – Response to Request for Additional Information

Reference:

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 (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," November 25, 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.

By letter dated November 25, 2005 (Reference 2), the NRC staff requested additional information regarding Entergy's probabilistic risk assessment studies that support the application for extended power uprate. Attachment 1 to this letter provides responses to the specific information requested.

There are no new regulatory commitments contained in this submittal.

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

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the original application, nor does it change Entergy's determination of no significant hazards consideration.

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 December 2, 2005.

Sincerely,

Jay K. Thayer
Jay K. Thayer
Site Vice President
Vermont Yankee Nuclear Power Station

Attachments (1)

cc: Mr. Samuel J. Collins (w/o attachment)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O 8 B1
Washington, DC 20555

USNRC Resident Inspector (w/o attachment)
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 43

Extended Power Uprate

Response to Request for Additional Information

Total number of pages in Attachment 1 (excluding this cover sheet) is 6.

NRC RAI APLA-A-1

Supplement 38, Attachment 1, page 9: Provide the engineering assessment that shows that the residual heat removal (RHR) and core spray (CS) pumps can operate at significantly reduced net positive suction head (NPSH) compared to the design NPSH, which is based on the results of tests conducted at Browns Ferry as described in NUREG/CR-2973. Have the conclusions of this engineering assessment been discussed with the pump manufacturer (Sulzer Bingham)? If so, does the pump manufacturer concur with the conclusions?

Response to RAI APLA-A-1

The discussion of pump operation at reduced levels of available NPSH in Supplement 38 was an indication of the margins provided against severe operational and mechanical degradation inherently available in the RHR and CS pump designs. However, for the risk evaluation of containment overpressure (COP) credit, the assumption is made that loss of containment results in loss of COP, which in turn results in failure of the pumps. This was a conservative assumption used to determine the risk impact of crediting COP on available NPSH.

The original engineering assessment for pump operation at reduced NPSH was performed by Tennessee Valley Authority engineers. The applicability of the results to Vermont Yankee Nuclear Power Station (VYNPS) RHR pumps was documented in a Yankee Atomic Electric Company memo written by R. Turcotte, dated January 13, 1993. This memo refers to a discussion with a nuclear pump expert at General Electric. There is no mention in the memo of any similar discussion with the pump manufacturer. The VYNPS RHR pumps and the Browns Ferry RHR pumps were purchased by General Electric under the same purchase order from Bingham (now Sulzer Pump Co.).

An evaluation of applicability of the results to VYNPS CS pumps is documented in Supplement 38 (Attachment 1, page 10). The VYNPS applicability assessments were not discussed with the pump manufacturer. As noted above, these results are not used in VYNPS deterministic evaluations of NPSH margin.

NRC RAI APLA-A-2

Supplement 38, Attachment 1, page 19 and Supplement 39, Attachment 1, pages 12 and 13: It is stated that EPRI TR-1009325 was used to determine the probabilities of containment pre-existing leakage. The NRC staff has not yet accepted this reference as a technical basis for granting permanent 15-year integrated leak rate test (ILRT) intervals. In fact, the Nuclear Energy Institute (NEI) submitted an updated version of this document for further staff review on October 26, 2005. The staff notes that the technical basis for containment leakage probabilities used to justify the one-time 15-year ILRT interval that was granted in VYNPS Amendment No. 227, dated August 31, 2005, was EPRI TR-104285, and that the containment leakage probabilities in this report are notably higher than those provided in EPRI TR-1009325. Either justify the use of EPRI TR-1009325 as an acceptable source of containment leakage probabilities, or reassess the change in core-damage frequency (CDF) caused by crediting containment accident pressure using containment leakage probabilities that are consistent with the recently granted one-time 15-year ILRT interval.

Response to RAI APLA-A-2

The basis for the VYNPS ILRT license change (license amendment 227) assumed that, for EPRI class 3a (small containment leakage), the leakage rate is 10La with a probability of 0.027. Similarly, for EPRI class 3b (large containment leakage), the leakage rate is 35La with a probability of 0.0027. These values are taken from NEI interim guidance for performing risk impact assessments in support of one-time extensions for Containment ILRT (issued in November 2001) and are considered conservative.

Because these leakage values are conservative, an update of the original EPRI evaluation (EPRI TR-104285) to assess the risk due to a revised containment leakage rate interval considered an expert elicitation process in the development of the probability of a large pre-existing containment leak. The results of this process are found in EPRI TR-1009325, which is currently undergoing a significant revision in which the expert elicitation results will be a sensitivity case in the analysis. However, the report also includes and references supporting documentation that compares the expert elicitation results with test data with good agreement.

The updated values are a better representation of leakage rates used in evaluating the one-time extension in the change for the ILRT interval. For example, these updated values considered a closer examination of previously observed containment leakage events that had been designated as failures, the potential risk benefits associated with additional containment inspections, and potential indirect containment monitoring techniques that would provide indications of a containment leak.

It is recognized that EPRI TR-1009325 has not yet been approved by the NRC. However, these updated values represent valuable improvements in evaluating containment leakage scenarios in risk sensitivity analyses.

NRC RAI APLA-A-3

Supplement 38, Attachment 1, page 20 and Supplement 39, Attachment 1, page 22: Provide the high confidence of low probability of failure (HCLPF) values used in the Seismic Margins Analysis (SMA) of VYNPS for the following: reactor coolant system piping, reactor vessel supports, safety relief valves (SRVs), and the containment.

Response to RAI APLA-A-3

HCLPF values were not computed for the reactor coolant system piping, reactor vessel supports, safety relief valves (SRVs), and the containment, because all of those components were screened out from the analyses.

From the VYNPS Individual Plant Examination on External Events (IPEEE), major components and equipment in the nuclear steam supply system (NSSS), which are located inside containment, are excluded from the scope of the Unresolved Safety Issue (USI) A-46 review. The NSSS primary coolant system (which includes reactor coolant system piping, reactor vessel supports, and SRVs) is screened out per EPRI NP 6041-SL.

The Seismic Class I structures (which includes the drywell) were also screened out since they are designed for a safe shutdown earthquake (SSE) greater than 0.1g. Mark I containments have undergone significant strengthening to resolve dynamic loading issues. The VYNPS torus, which was originally designed for seismic loads, has thus been significantly strengthened. Because of this, the IPEEE reviewers determined that significant margin had been added to the design of the torus and that the torus should be screened out.

NRC RAI APLA-A-4

Supplement 38, Attachment 1, page 20 and Supplement 39, Attachment 1, page 22: Could a fire simultaneously cause a stuck-open relief valve and a failure of the containment isolation (CI) system?

Response to RAI APLA-A-4

There is no postulated fire that could simultaneously cause a stuck-open relief valve and a failure of containment isolation. The following information is taken from the VYNPS IPEEE, Rev. 2 on Internal Fires Analysis (2004):

> Power cables for each safety relief valve (SRV) circuit are isolated in grounded, steel raceway (conduit) from the control room floor, through the cable vault and reactor building, through the drywell electrical penetration to the associated SRV. The steel conduit is dedicated to each SRV power solenoid (SOV) circuit and no other power source cables are located within these conduits. The power cables for the SRVs are IEEE 383 qualified. Given the SRV circuit design and the grounded, steel conduit isolation in the reactor building and cable vault, the likelihood of a fire-induced hot short causing spurious opening of an SRV is judged to be very remote and is not evaluated further. Fire damage to SRV power cables in the reactor building and cable vault is assumed to fail the associated SRV in the de-energized, closed position. The power cables for SRV-71A and SRV-71B are routed separately from SRV-71C and SRV-71D cables in the reactor building.
>
> An automatic depressurization system (ADS) inhibit switch is provided in the control room for manual blocking of a postulated fire-induced ADS signal which could cause the SRVs to open. The inhibit switch interrupts both the positive and negative legs of each SRV power circuit. The inhibit switch enclosure and downstream conduit leading to the control room to cable vault penetration are protected with a 1-hour rated fire barrier. Given the inhibit switch design and the rated fire barrier protection, the likelihood of a fire-induced hot short causing a spurious opening of an SRV is judged to be very remote for control room fires. Section 4.10.3 addresses LOCA events due to SRV opening and failure to re-close in the evaluation of control room fires.

NRC RAI APLA-A-5

Supplement 39, Attachment 1, general: Is the overall intent of the risk evaluation of the proposed containment overpressure credit to provide a sensitivity analysis that investigates modeling uncertainty in the baseline post-EPU PRA? The NRC staff notes that Supplement 38 indicates no overpressure credit is required using realistic assumptions. Hence, there should be no changes between the pre-EPU and post-EPU PRA models with respect to their treatment of the proposed overpressure credit.

Response to RAI APLA-A-5

The overall intent of the risk evaluation of the proposed COP credit is to provide a very specific sensitivity analysis on this one aspect (i.e., need/no need for COP). Because no COP is required using realistic assumptions, this analysis (needing COP) represents a special case and falls outside the baseline post-EPU PSA. There is no PRA modeling change between pre-EPU and post-EPU with respect to COP. The comparison is drawn only for insight purposes and does not supersede the deterministic analyses which will become the new licensing basis for VYNPS.

NRC RAI APLA-A-6

Supplement 39, Attachment 1, general: Does the change in CDF only consider the impact of the proposed overpressure credit, or does it also include the impact of other changes resulting from the proposed EPU (e.g., shorter operator times due to higher decay heat)?

Response to RAI APLA-A-6

The change in CDF in Attachment 1 of Supplement 39 only considers the impact of the proposed COP credit. The base case was the post-EPU case with realistic inputs and does not require crediting for COP. The evaluation provided in Supplement 39 was only provided as a sensitivity analysis. It does not alter the base EPU evaluation.

NRC RAI APLA-A-7

Supplement 39, Attachment 1, page 13: It is stated that containment integrity (Event IP) is considered when the hardened torus vent is being used (Event VT) to prevent over-pressurization failure of the containment following a loss of torus cooling (Event TC). It is difficult to interpret the event tree logic (e.g., the large loss-of-coolant accident (LOCA) event tree) in the context of this statement since Event IP appears before Event VT. To help clarify the NRC staff's understanding of the modeling approach taken, provide a narrative explanation of each core-damage sequence in the large LOCA event tree.

Response to RAI APLA-A-7

In conjunction with development of the COP risk assessment (i.e., Supplement 39), additional thermal-hydraulic analyses were performed using the Modular Accident Analysis Program (MAAP) computer code in order to evaluate operator response timing. An error was discovered when a review of MAAP computer runs determined that operator action to control torus venting is ineffective in controlling containment pressure to preclude NPSH concerns for low pressure coolant injection (LPCI) and core spray pumps. Therefore, operator action AINPSH, "Operator

Fails to Control Vent and LP Fails due to Loss of NPSH", which involved the potential failure by the operator to adequately control torus venting such that NPSH is lost and ECCS pump failure is assumed to occur. In conclusion, containment integrity is not considered in relation to operation of the hardened torus vent (Event VT) to prevent over-pressurization failure of the containment following a loss of torus cooling (Event TC). The engineering report (i.e., Supplement 39, Attachment 1) has been revised to reflect this model change, and the referenced statement was deleted.

NRC RAI APLA-A-8

Supplement 39, Attachment 1, page 14: If the containment is not intact (Event IP occurs), why is it possible to credit alternative injection and containment overpressure (COP) control (Event AI)?

Response to RAI APLA-A-8

The statement on page 14 (Supplement 39, Attachment 1) was in error. As stated above in response to RAI APLA-A-7, the VYNPS PRA model was changed, deleting the use of COP control in top event AI (Alternate Injection). The engineering report provided in Supplement 39 has been revised in this regard.

NRC RAI APLA-A-9

Supplement 39, Attachment 1, page 18 and Tables 3.2A and 3.3: On page 18, it is stated that CONFIG#1 represents the risk when the COP is not available and CONFIG#2 represents the risk when the COP is available. However in Tables 3.2A and 3.3, the CDF associated with CONFIG#1 is lower than for CONFIG#2. Please clarify. Also, note that in Table 3.3, the total CDF for CONFIG#1 is incorrect (typographical error).

Response to RAI APLA-A-9

The statement on page 18 of the engineering report provided in Supplement 39 was revised to more clearly state that CONFIG#1 represents the risk when COP is not necessary to satisfy RHR and CS pump NPSH requirements, and that CONFIG#2 represents the risk when the COP is necessary to satisfy RHR and CS pump NPSH requirements.

The typographical error in Table 3.3 (the total CDF for CONFIG#1) has been corrected in the revised engineering report.

NRC RAI APLA-A-10

Supplement 38, Attachment 1, page 18: It is stated that the only difference between the model cases lies in End state Bin IIV. However, Table 3.2A indicates that End state Bins ID, IIIC, IVA, and IC also change. Please clarify.

Response to RAI APLA-A-10

The statement on page 18 of the engineering report provided in Supplement 39 is incorrect. The engineering report has been revised, deleting the statement which referred to end state bin IIV.

The minor decrease in end state bins IIV and IVA (~2E-11) was due to capture of some existing sequences for CONFIG#1 in end state bins ID and IC for CONFIG#2. This occurred when the end state binning rules were revised to reflect the change in success criteria used for CONFIG#2 (i.e., COP is necessary to satisfy RHR and CS pump NPSH requirements). End state bins ID, IIIC and IC reflect the increase in CDF between CONFIG#1 and CONFIG#2. The engineering report provided in Supplement 39 has been revised for completeness and now includes split fraction (SF) sequence information relative to end state bin ID for CONFIG#1 and CONFIG#2 (i.e., Tables 4.3.C and 4.4.C, respectively), in addition to the existing SF sequence information for end state bins IC and IIIC.