December 1, 2003

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

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ALTERNATIVE SOURCE TERM REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MC0253)

Dear Mr. Kansler:

By letter dated July 31, 2003, as supplemented on October 10, 2003, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensees) submitted an amendment application to incorporate an Alternative Source Term methodology into the Vermont Yankee Nuclear Power Station licensing basis. In order for the staff to complete its review of this application, we are requesting a response to the questions provided in the enclosure. Please provide your response to these questions by December 30, 2003.

If you have any questions please contact me at (301) 415-3016.

Sincerely,

/RA/

Robert M. Pulsifer, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION
ALTERNATE SOURCE TERM
ENTERGY NUCLEAR VERMONT YANKEE, LLC
VERMONT YANKEE NUCLEAR POWER STATION
DOCKET NO. 50-271

1. In the application dated July 31, 2003, on page 54 of Attachment 5, it is indicated that seismic criteria were used to evaluate the ruggedness of the alternative leakage treatment pathways and boundary piping, equipment, and supports. Describe the seismic criteria used.

2. Provide a list of the outliers and their resolution.

3. Please provide a copy of reference 28, specified in your July 31, 2003 submittal on page 51 of Attachment 5.

4. The values of certain containment volumes used in the pH calculations (page 9 of the analysis) appear to differ from the design values given in the Updated Final Safety Analysis Report (UFSAR), Revision 17, Table 1.7.4. These values were for suppression pool volume, drywell volume, and torus airspace. Please explain the values selected for the pH calculations.

5. The amount of water added with the sodium borate from the Standby Liquid Control System to the suppression pool is not specified (i.e., the borate concentration). This additional water may affect the pH calculation. Please identify the amount of water added to the suppression pool with the borate, and discuss its potential effect on the pH calculations.

6. In the Safety Assessment (Attachment 5) for this amendment request, on page 8 it states that an assumed post-isolation unfiltered inleakage rate equal to the pre-isolation unfiltered intake rate (3700 cubic feet per minute(cfm)) into the control room is used for the analyses. No credit was taken for manually placing the control room ventilation system into recirculation mode after the accident. Control room isolation is only achieved through manual initiation. In effect, these assumptions imply no credit for control room isolation after a design basis accident (DBA).

   a. Have you performed control room envelope inleakage testing? If so, what was the result, and how does that compare to the assumption of 3700 cfm of unfiltered inleakage?

   b. Have you determined how the dose results of each of the postulated DBAs would be affected by assuming manual control room isolation occurs with a lesser unfiltered inleakage rate than currently assumed? A greater unfiltered inleakage rate?

7. You propose to take credit for drywell spray removal of iodine. Do the drywell sprays meet requirements given in Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A, General Design Criteria 41, 42 and 43 with regard to containment atmosphere cleanup systems? Can the drywell sprays perform their safety function with

a single failure? Have you verified the capability of the system to deliver the assumed spray flow?

8. With respect to Attachment 5, please provide a detailed description of all inputs and assumptions for all relative concentration (X/Q) values not previously approved by the U.S. Nuclear Regulatory Commission (NRC). This appears to include the exclusion area boundary ground level reactor building (RB) bypass and siding, and stack 2-8 and 8-24 hour X/Q values; the low population zone (LPZ) ground level main steam isolation valve (MSIV) and ground level RB bypass and siding X/Q values; and the control room X/Q values for the loss-of-coolant accident (LOCA), main steam line break (MSLB), and fuel handling accidents. Note 3 of Table 2-6 states that control rod drop accident ground level release LPZ X/Q values are documented in the UFSAR. Were these values previously approved by the NRC? If so, please provide a reference citation.

9. For the LOCA LPZ estimates, why is there a difference between the ground level MSIV and RB siding and bypass X/Q values? Is this only because the AEOLUS-3 methodology was used in some of the calculations and the SKIRON-II methodology used when making other estimates? Provide a description of the differences between the AEOLUS-3, SKIRON-II and Regulatory Guide (RG) 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” methodologies and the expected impact of these differences on resultant estimated X/Q values.

10. What are the specifics of your MSLB puff calculation and calculational inputs and assumptions used in the comparison calculations with the puff methodology described in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants?”

11. What is the distance between the turbine building release location and the control room ventilation intake location?

cc:

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