December 16, 2003

Mr. William K. Sherman
Vermont Department of Public Service
112 State Street
Drawer 20
Montpelier, VT 05620-2601

Dear Mr. Sherman:

I am responding to your letter dated August 8, 2003, to the U.S. Nuclear Regulatory Commission (NRC), expressing concerns over a proposed modification to the Vermont Yankee Nuclear Power Station (VYNPS) design bases. Your concerns focused specifically on the licensee's analysis of no significant hazards consideration (NSHC) in their July 31, 2003, application.

This amendment requested a revision to the design basis of the VYNPS to support the application of an alternate source term methodology. You requested assistance from the NRC in understanding issues regarding the licensee's NSHC determination. You also asked for information on the review criteria the NRC staff uses, and acceptability of the licensee's amendment request for alternate leakage treatment pathway for main steam isolation valve leakage, and the standby liquid control system credit.

The NRC staff's response to your questions is enclosed. We have provided information on NSHC determinations, as we discussed with you in our telephone conference on August 21, 2003. We are also providing background and review criteria for technical issues raised in your letter. The NRC staff will document whether or not the licensee's proposed amendment is acceptable in our safety evaluation (SE) for the requested amendment. I trust that this information will address your concerns.

Sincerely,

/RA/

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

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

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Sincerely,

/RA/
Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
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Responses to Questions from
Vermont Department of Public Service
Proposed Alternative Source Term Amendment for Vermont Yankee Nuclear Power Station

Question 1 (a): Does the agency have criteria or guidance for how to interpret the word "significant" in 10 C.F.R. §50.92(c)(1)?

Answer: The U.S. Nuclear Regulatory Commission (NRC) staff would like to point out that, as discussed in the *Federal Register* (FR) notice, "Final Procedures and Standards on No Significant Hazards Consideration," dated March 6, 1986 (51 FR 7744), the no significant hazards consideration (NSHC) is a procedural standard and not a safety standard. Any conclusion the NRC staff may reach in determining whether or not a proposed change involves a significant hazards consideration is only used to determine whether a hearing, if one is requested, would be held before issuance of the associated amendment, if granted, or whether it could be held after issuance. As such, the staff does not apply the NSHC criteria in making its safety finding of adequate protection of the public health and safety. The staff makes its safety finding based on whether or not a proposed change meets applicable design and operational requirements.

The NRC has issued Regulatory Issue Summary (RIS) 2001-22, "Attributes of a Proposed No Significant Hazards Consideration Determination," dated November 20, 2001, which is available electronically at the NRC's website in the Agencywide Documents Access and Management System (ADAMS) under accession number ML011860215. This RIS provides guidance on preparing an NSHC analysis for a license amendment request. The NRC staff, in Attachment 1 to the RIS, discussed criteria for how to determine if a proposed change is "significant." The RIS states that if a value of probability or consequence of an event continues to meet the current licensing basis or applicable ranges of values in guidance, the change would generally not be considered significant.

Question 1 (b): How can we know whether the proposed MSIV [main steam isolation valve] leakage change is significant when we haven't been presented with a comparable before and after design basis accident calculation for comparison?

Answer: Whether a change involves a significant increase in the probability or consequences of an accident previously evaluated is determined on a case-by-case basis. The significance determination should include a comparison of the probability or consequences of an accident before the change to that after the change. As discussed above, however, the NRC staff generally considers that if the probability or consequences of an event related to a proposed change remains within those in applicable guidance, then the change is not considered significant. As such, the staff considers that detailed comparison of calculation or analytical results is not necessary to reach the NSHC finding.

Enclosure

Question 2 (a): We believe that the definition of Engineered Safety Feature (ESF) is: a feature provided to mitigate the consequences of design basis or loss-of-coolant accidents (see SRP 6.1.1). Since Entergy [Entergy Nuclear Vermont Yankee, LLC,] is claiming credit for the alternate leakage treatment pathway to mitigate the radiological consequences of design-basis accidents, it appears the alternate leakage treatment pathway meets the definition of an ESF. Is that a correct interpretation?

Answer: Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 100 requires structures, systems and components necessary to assure the capability of the plant to mitigate the consequences of design-basis accidents to be designed to remain functional during and after a safe shutdown earthquake (SSE). Appendix F of the VYNPS Updated Final Safety Analysis Report specifies that ESF components are to function following a design basis loss-of-coolant accident (LOCA) and are to withstand the most severe forces and environmental effects resulting from the accident. Appendix A of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors"(ADAMS accession number ML003716792), allows credit for the components and piping systems used in the release path if they are capable of performing their safety functions during and following an SSE.

Entergy proposed to use the alternate leakage treatment (ALT) method presented in the General Electric (GE) licensing topical report NEDC-31858P-A, Revision 2, "BWROG [Boiling Water Reactor Owners Group] Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," to assure that the ALT pathway for MSIV leakage is suitable for its intended safety function. The NRC staff's safety evaluation (SE) approving NEDC-31858P-A, Revision 2, is dated March 3, 1999 (ADAMS accession number ML010640286).

The MSIV leakage ALT pathway will be required to prevent a direct release of fission products that could leak through the closed MSIVs after a LOCA. Based on the criteria cited above, the piping and components of the MSIV leakage ALT pathway will be evaluated as ESFs.

Question 2 (b): While it is clear that the alternate leakage treatment pathway was not designated as an ESF in the original design basis, is it appropriate to employ Section C.5.1.4 of Regulatory Guide 1.183 in the review of the alternate leakage treatment pathway?

Answer: Inasmuch as the proposed ALT relates to implementation of an alternate source term (AST), RG 1.183 would be applied. Section C.5.1.4, "Applicability of Prior Licensing Basis," of RG 1.183, notes that prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Additionally, Section C.5.1.4 indicates that characteristics of the AST may be incompatible with analysis assumptions and methods currently reflected in the facility's design-basis analysis. Independent of the ALT pathway designation, Section C.5.1.4 is applicable to general considerations employed during review of the AST. Specifically, if the NRC staff finds that new or unreviewed issues are created by the implementation of the AST, review of staff positions relied upon in approving the initial issuance of the license may be warranted.

Question 2 (c): Standard Review Plan [SRP] (NUREG-0800) 6.7, July 1981, Rev. 2, is titled, *Main Steam Isolation Valve Leakage Control System (BWR)*. Is this the latest revision of SRP 6.7?

Answer: Revision 2 is the latest revision in use. The NRC staff developed a draft Revision 3, dated April 1996, which discusses the alternate leakage treatment pathway. However, Revision 3 has not been finalized.

Question 2 (d): It appears that SRP 6.7 should be applicable to the review of the proposed alternate leakage treatment pathway. Is that a correct interpretation?

Answer: SRP 6.7 allows a licensee to propose alternative methods to comply with specific regulations. The NRC staff's SE for NEDC-31858P-A, Revision 2, indicated that the BWROG report is acceptable for direct reference by plant-specific submittals on MSIV leakage issues. The staff also indicated that use of NEDC-31858P-A, Revision 2, is subject to the conditions and limitations stated in the staff's SE. Entergy proposed to use the ALT method presented in the BWROG report as an alternative to SRP 6.7. The acceptability of this proposal will be addressed in our final SE.

Question 2 (e): Entergy claims credit for the seismic adequacy of the alternate leakage treatment pathway in the Safety Assessment. According to SRP 6.7, the acceptance criteria for seismic adequacy is based on meeting position C.1 of Regulatory Guide 1.29. Is this a correct interpretation, and if so, does the alternate leakage treatment pathway proposed by Entergy meet the requirements of position C.1 of Regulatory Guide 1.29?

Answer: As discussed in the answer to question 2.(d), in lieu of the guidance in SRP 6.7, the NRC staff will base its review on the approved GE licensing topical report, NEDC-31858P-A, Revision 2. The staff's review of the amendment request is not complete at this time; therefore, it is premature to make a determination if applicable acceptance criteria are met. We will address the adequacy of the licensee's request in our final SE.

Question 2 (f): Position C.1 of Regulatory Guide 1.29 states, in part, "The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to activities affecting the safety-related functions." Is it a correct interpretation that the alternate leakage treatment pathway must meet the requirements of Appendix B to 10 CFR Part 50, and if so, is this requirement met?

Answer: Position C.1 of RG 1.29 provides acceptance criteria for SRP 6.7. As discussed above, in lieu of the guidance in SRP 6.7, the NRC staff will base its review on the approved GE licensing topical report, NEDC-31858P-A, Revision 2. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 establishes quality assurance requirements that are applicable to all activities affecting the safety-related functions of structures, systems and components, including the ALT pathway. Moreover, the BWROG report specifies that licensees should provide assurance for the reliability of the ALT pathway. The NRC staff's review of the amendment request is not complete at this time; therefore, it is premature to make a determination if applicable requirements are met. We will address the adequacy of the licensee's request in our final SE.

Question 3 (a): Regulatory Guide 1.183, Section C.5.1.2 states, in part: "The single active component failure that results in the most limiting radiological consequences should be assumed." Does this section apply to Entergy proposed crediting of the SLC [standby liquid control] system to maintain suppression pool pH?

Answer: The SLC system will be reviewed against the guidance in Section C.5.1.2, "Credit for Engineered Safeguard Features," of RG 1.183 as part of the plant design. Safety systems meant to control fission products released to the environment following postulated accidents should have suitable redundancy and reliability to assure that their safety functions can be accomplished. The NRC staff is currently evaluating the use of the SLC system for pH control.

Question 3 (b): Is it correct that the single failure referred to above is interpreted as a non-mechanistic failure of "[t]he single active component failure that results in the most limiting radiological consequences?"

Answer: It is unclear as to what "single failure" this question refers. As indicated in the answer to question 3(a), the NRC staff is currently evaluating the use of the SLC system. Specific interpretation and application of Section C.5.1.2 of RG 1.183 is currently being evaluated by the NRC staff on a generic basis. The staff will determine whether, and to what extent, the SLC system is subject to this guidance. General guidance on single failure is provided in SECY-77-439, "Single-Failure Criterion," dated August 17, 1977, and RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plants Protection Systems" (ADAMS accession number ML003740182). SECY-77-439 provides definitions for active and passive failures in fluid systems, but also notes there are challenges in the application of the single-failure criterion to fluid systems. We will address the adequacy of the licensee's request in our final SE.

Question 3 (c): In the Safety Assessment, at pp. 3-4, Entergy identifies a single failure of the control room keylocked-switch and associated logic which could defeat the function of the SLC system. It proceeds to dismiss this single failure based on low failure rates. Is dismissal of the single failure based on low failure rates an acceptable method of meeting the single-failure criteria identified by Regulatory Guide 1.183, section C.5.1.2, and if so, what is the agency's basis for accepting this methodology?

Answer: The Institute of Electrical and Electronical Engineer's Standard 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," which the NRC endorsed in RG 1.53, indicates that the performance of a probabilistic assessment of safety systems may be used to demonstrate that certain postulated failures need not be considered in the application of the single-failure criterion. A probabilistic assessment cannot be used in lieu of the single-failure analysis. However, reliability analysis, probability assessment, operating experience, and engineering judgment, or a combination thereof, may be used to establish a basis for excluding a particular failure from the single-failure analysis.

The NRC staff's review of the amendment request is not complete at this time; therefore, it is premature to make a determination if the low failure rates presented by Entergy provide an acceptable basis for exclusion of any failure from the single-failure analysis. We will address the adequacy of the licensee's request in our final SE.