August 31, 2005

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

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF INTEGRATED LEAK RATE TEST INTERVAL (TAC NO. MC4662)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 227 to Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated October 5, 2004, as supplemented on April 22, 2005.

The amendment revises Technical Specification 6.7.C "Primary Containment Leak Rate Testing Program," to allow a one-time extension to the 10-year interval for performing the next Type A containment integrated leak rate test (ILRT). Specifically, the change would allow the test to be performed within 15 years from the last ILRT which was performed in April 1995.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

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

Docket No. 50-271

Enclosures: 1. Amendment No. 227 to License No. DPR-28
2. Safety Evaluation

cc w/encls: See next page

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Vermont Yankee Nuclear Power Station

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ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 227
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensee) dated October 5, 2004, as supplemented on April 22, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 227, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 31, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
265	265

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE NO. DPR-28

ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated October 5, 2004, as supplemented on April 22, 2005, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications (TSs). The proposed amendment would revise TS 6.7.C "Primary Containment Leak Rate Testing Program," to allow a one-time extension to the 10-year interval for performing the next Type A containment integrated leak rate test (ILRT). Specifically, the change would allow the test to be performed within 15 years from the last ILRT which was performed in April 1995.

The supplement dated April 22, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 21, 2004 (69 FR 76492).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. VYNPS TS 6.7.C requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with one exception (discussed in the next paragraph). This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at VYNPS have been successful, so the current interval requirement would normally be 10 years. However, by letter dated October 4, 2002, the licensee requested a one-time extension of the test interval to 10.6 years. On June 2, 2003, the Nuclear Regulatory Commission (NRC or Commission) staff granted this request via Amendment No. 215. Based on the changes approved in Amendment No. 215, TS Section 6.7.C, currently states, in part, that:

> The first Type A test after the April 1995 Type A test shall be performed no later than November 2005.

The proposed amendment would modify the existing exception from the guidelines of NEI 94-01 regarding the Type A test interval. The exception would allow ILRT testing within 15 years from the last ILRT which was performed in April 1995. Specifically, the sentence quoted above in TS 6.7.C would be modified to state:

> The first Type A test after the April 1995 Type A test shall be performed no later than April 2010.

The Type B and C local leak rate tests (LLRTs), including their schedules, are not affected by this amendment request.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 Risk Impact

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the application dated October 5, 2004. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

Although the plant currently has a one-time interval of 10.6 years, the NRC staff will conservatively use the generic 10-year interval as the basis for this evaluation.

The basis for the 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during LLRTs based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized water reactor and boiling water reactor (BWR) representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak was present. Since the Option B rulemaking was completed in 1995, the NRC staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The following comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The NRC staff reached the following conclusions based on review of the licensee's analysis associated with extending the Type A test frequency:

1. Given the change from a three-in-10-year test frequency to a one-in-15-year test frequency, the increase in the total integrated plant risk is estimated to be approximately 0.01 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be 6.9×10^{-9} per year based on the internal events PRA, and 8.8×10^{-8} per year including both internal and external events. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting

large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8} per year. The NRC staff concludes that increasing the Type A interval to 15 years results in a very small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of approximately 0.1 percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.2 Inservice Inspection (ISI) for Primary Containment Integrity

VYNPS is a General Electric BWR-4 plant with a Mark I design primary containment. The containment consists of three primary interconnected structures: a drywell which encloses the reactor vessel, a pressure suppression chamber (torus) which stores a large volume of water, and a connecting vent system between the drywell and the torus. The primary containment design includes several different types of penetrations (e.g., piping penetrations, electrical/instrumentation penetrations, and personnel and equipment hatches and locks).

The leak rate testing requirements (ILRT and LLRTs) of Option B of Appendix J to 10 CFR Part 50, and the containment ISI requirements mandated by 10 CFR 50.55a complement each other in ensuring the leak-tightness and structural integrity of the containment. Based on its review of Type A test interval extension applications for other plants, and related plant operating experience, the NRC staff has identified several areas that the licensee should address with regard to the ISI of the containment when requesting an ILRT interval extension. These areas are addressed in safety evaluation (SE) Sections 3.2.1 through 3.2.4.

3.2.1 ISI Program Methods and Schedule Used to Identify Containment Degradation

The licensee should provide a description of the ISI program methods and schedule used to assess the general condition of the containment and to detect evidence of degradation that may affect structural integrity or leak tightness.

In the application dated October 5, 2004, the licensee provided a general description of the VYNPS ISI program and stated that the program contains detailed ISI requirements for Class MC components (pressure-retaining metallic components) in accordance with

10 CFR 50.55a(b)(2)(vi) and (ix), and the 1998 Edition of ASME Code Section XI through the 2000 Addenda. Under this program, VYNPS performs Category E-A examinations in accordance with Table IWE-2500-1. Included in the general visual examinations are the interior and exterior pressure retaining boundary (Item E1.10), accessible surface areas (Item E1.11), and moisture barriers (Item E1.30). The general visual examinations focus on coating flaws (such as cracking, peeling, flaking, blistering, rusting, and discoloration), and any mechanical damage, pitting and arc strikes. Any indications of degradation identified are to be recorded and evaluated.

In Attachment 2 of the supplement dated April 22, 2005, the licensee provided the engineering standard used to perform the visual examination of the interior and exterior containment surfaces at VYNPS. In general, the standard contains acceptance criteria derived from various American Society for Testing and Materials (ASTM) documents for determining the condition of coated and uncoated surfaces (e.g, extent of corrosion, blistering, flaking, cracking, etc.). The standard states the inspection acceptance criteria have been evaluated and have been determined to not deter or compromise the structural integrity of the primary containment pressure boundary. The standard states that if the inspection results indicate the need for further evaluation, a detailed visual, VT-1, examination or additional examinations (e.g., ultrasonic testing) will be performed.

In Attachment 3 of the supplement dated April 22, 2005, the licensee provided a table showing the ISI program schedule for the containment visual examinations. The table includes a listing of the specific items inspected for the torus and torus penetrations, the drywell and drywell penetrations, and the vent system. The schedule in the table indicates that the inspections were last performed during refueling outage (RFO) 24, which occurred in Spring 2004, and that the next inspections are scheduled for RFO 27 (Fall 2008) and RFO 29 (Fall 2011). The last ILRT was performed in April 1995 and the proposed amendment would allow the next ILRT to be performed in April 2010 (RFO 28). Therefore, the containment ISI schedule complements the ILRT by providing more frequent opportunities to detect evidence of containment degradation.

Based on the review of the licensee’s submittals, the NRC staff finds that the VYNPS ISI program methods and schedules are acceptable for assessing the general condition of the containment and for detecting evidence of degradation that may affect containment structural integrity or leak tightness.

3.2.2 Augmented Examinations

Containment interior and exterior surface areas likely to experience accelerated degradation and aging require augmented examinations. Subsection article IWE-1240 of Subsection IWE of the ASME Code requires the identification of the surface areas requiring augmented examinations. Paragraph IWE-1241 provides the selection criteria for those areas requiring augmented examinations.

In the application dated October 5, 2004, the licensee indicated that based on the ISI examinations performed, no containment surface areas currently require augmented examination in accordance with IWE-1241. The NRC staff finds that the licensee has adequately addressed the concern related to augmented examinations for VYNPS.

3.2.3 Examination and Testing of Seals, Gaskets, and Bolted Connections

For the examination of penetration seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary (Examination Categories E-D and E-G), relief from the requirements of the ASME Code had been requested by some nuclear plant licensees. As an alternative, these licensees proposed to examine the containment components during the leak-rate testing of the primary containment. Licensees should indicate if relief has been requested for these items and provide a description of how these items are examined and the schedule for the examinations.

The licensee's application dated October 5, 2004, stated that there are no relief requests in effect for the containment ISI program. In the supplement dated April 22, 2005, the licensee stated that the ASME Code of Record for VYNPS does not contain Categories E-D or E-G. However, containment bolted connections, Table IWE-2500-1, Category E-A, Item E1.11, are examined in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(H) which requires that:

> Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 examination method. Flaws or degradation identified during the performance of a VT-3 examination must be examined in accordance with the VT-1 examination method. The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

The NRC staff finds that the VYNPS ISI program provides reasonable assurance that any degradation of bolted connections will be evaluated such that corrective actions will be taken as needed to maintain the primary containment pressure boundary.

3.2.4 Integrity of Stainless Steel Bellows

NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," was issued to alert licensees of problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellow were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem. Licensees should discuss the applicability of this issue to their facility and, if applicable, provide information regarding inspection and testing of the bellows, and how such behavior has been factored into the risk assessment submitted to support the ILRT interval extension request.

In the supplement dated April 22, 2005, the licensee stated that the expansion bellows at VYNPS utilize a single-ply design. As such, the Type B test are capable of detecting through-wall defects. The NRC staff finds that the concerns of Information Notice 92-20 are not applicable to VYNPS and no ISI of these bellows is required.

3.3 Technical Evaluation Conclusion

Based on the considerations in SE Sections 3.1 and 3.2, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. In an e-mail to the NRC Project Manager, Mr. Richard Ennis, dated July 25, 2005, the State official, Mr. William Sherman, provided the following two comments regarding the proposed amendment:

1) As you know, we are questioning the proposed granting of containment overpressure credit to demonstrate adequate NPSH [net positive suction head] for safety related cooling pumps for power uprate. It is obvious to us that if such overpressure credit is granted, it would be desirable to test containment integrity. Therefore, we believe it undesirable to grant the requested extension if overpressure credit is going to be granted.

2) Also, we would like to understand the root cause of the recent discovery of loss of containment integrity at the Fitzpatrick nuclear plant, a sister BWR plant to Vermont Yankee.

In a phone call on August 10, 2005, Mr. Sherman told Mr. Ennis that he no longer considered the second comment to be an issue and that the NRC staff need only address the first comment.

Containment overpressure relates to the crediting of a portion of the calculated containment accident pressure in the calculation of the NPSH for the emergency core cooling system pumps. The VYNPS design/licensing basis does not currently include credit for containment overpressure. The licensee has requested a change to the VYNPS design/licensing basis as part of a separate amendment request that is currently under NRC staff review (i.e., extended power uprate (EPU) amendment request dated September 10, 2003). The NRC staff has not made any determination at this time whether the proposed EPU amendment request for VYNPS (and the associated use of containment overpressure credit) is acceptable. As such, the use of containment overpressure credit was not evaluated as part of the ILRT interval extension amendment request. The NRC staff will include an ILRT test interval of 15 years as one of the inputs in the risk evaluation that will be performed as part of the EPU amendment review of the containment overpressure issue.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 76492). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in

10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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R. Ennis

Date: August 31, 2005