UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 13, 2009

Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
P.O. Box 250
Governor Hunt Road
Vernon, VT 05354

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: RELOCATION OF REACTOR BUILDING CRANE TECHNICAL SPECIFICATION (TAC NO. MD9725)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 239 to Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated September 22, 2008, as supplemented by letter dated October 31, 2008.

The amendment would relocate the contents of the Vermont Yankee (VY) Technical Specification (TS) relating to the Reactor Building crane to the VY Technical Requirements Manual.

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,

James Kim

James Kim, Project Manager
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures:
1. Amendment No.239 to License No. DPR-28
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 239
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 September 22, 2008, as supplemented by letter dated October 31, 2008, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 239 , are hereby incorporated in the license. Entergy Nuclear Operations, Inc. 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: July 13, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 239

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
235	235
236	236

E. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts .30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20. Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1912 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 239are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

C. Reports

Entergy Nuclear Operations, Inc. shall make reports in accordance with the requirements of the Technical Specifications.

D. This paragraph deleted by Amendment No. 226.

E. Environmental Conditions

Pursuant to the Initial Decision of the presiding Atomic Safety and Licensing Board issued February 27, 1973, the following conditions for the protection of the environment are incorporated herein:

3.12 LIMITING CONDITIONS FOR OPERATION

F. Fuel Movement

The reactor shall be shut down for a minimum of 24 hours prior to fuel movement within the reactor core.

G. Deleted

4.12 SURVEILLANCE REQUIREMENTS

F. Fuel Movement

Prior to any fuel handling or movement in the reactor core, the licensed operator shall verify that the reactor has been shut down for a minimum of 24 hours.

G. Deleted

3.12 LIMITING CONDITIONS FOR OPERATION

H. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is stored in the spent fuel pool, the pool water temperature shall be maintained below 150°F.

4.12 SURVEILLANCE REQUIREMENTS

H. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be recorded daily. If the pool water temperature reaches 150°F, all refueling operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 239 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 application dated September 22, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082700458), as supplemented by letter dated October 31, 2008 (ML083110414), Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station (Vermont Yankee or VY) Technical Specifications (TSs). The amendment changes would relocate the contents of the VY TS relating to the Reactor Building crane to the VY Technical Requirements Manual (TRM). The supplemental letter dated October 31, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 18, 2008 (73 FR 68454).

2.0 REGULATORY EVALUATION

2.1 Regulatory Discussion

Content of Technical Specifications

Section 182a of the Atomic Energy Act of 1954, as amended (the Act) requires applicants for nuclear power plant operating licenses to include the TSs as part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." The regulation requires that the TSs include items in specific categories, including: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in the TSs.

The four criteria defined by 10 CFR 50.36(c)(2)(ii) for determining whether particular items are required to be included in the TS LCOs, are as follows:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The DBAs and transient analyses discussed in Criteria 2 and 3 include any design basis event described in the Safety Analysis Report (SAR), not just those events described in the accident analysis chapter. The "initial conditions" captured under Criterion 2 should not be limited to only "process variables" assumed in the safety analyses, but should also include certain active design features and operating restrictions needed to preclude unanalyzed accidents. In this context, "active design features" include only those design features under the control of operations personnel (i.e., licensed operators and personnel who perform control functions at the direction of licensed operators). Therefore, if the TS involves physical, designed-in features that prevent operations staff from immediately exceeding the assumptions in the bounding analysis in the course of operations, the TS would not meet Criterion 2 and could be relocated to the SAR or other similarly controlled document.

2.2 System Description

The Reactor Building crane is a 110-ton capacity overhead bridge crane that provides services for the reactor and refueling area. The crane handles new and spent fuel, in-core detectors, a large segmented concrete plug in the refueling level floor, the drywell head, the reactor vessel head, the segmented pool plugs, and the spent fuel shipping cask.

The Reactor Building crane was modified in 1976 by replacing the original trolley with one that has a dual load path on the main hoist when used for cask handling operations. The design of the new trolley satisfies the criteria for dual load path or "single-failure-proof" cranes, and, with issuance of Amendment 29 to the VY operating license on January 28, 1977, the NRC staff accepted the crane as "single-failure-proof."

In addition to the hardware improvements implemented to modify the crane to "single-failure-proof," the licensee implemented a number of other improvements to enhance conformance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." These improvements included:

- revising maintenance procedures to define safe load paths for major loads.
- revising procedures to include training and qualification requirements for crane operators, sling selection criteria, crane inspections prior to use, and supervisory oversight of heavy lift operations.
- procurement of special lifting devices and performance of periodic non-destructive examinations to monitor the condition of lifting devices.

The NRC staff accepted these improvements through a safety evaluation transmitted by letter dated June 27, 1984.

3.0 TECHNICAL EVALUATION

In the letter dated September 22, 2008, the licensee stated that the proposed change is to relocate the sections of the TS relating to the Reactor Building crane to the VY TRM. The licensee proposed relocating the Reactor Building crane sections of the TS to the TRM because the Reactor Building crane is neither within the scope of equipment included in the "Standard Technical Specifications, General Electric Plants, BWR/4," NUREG-1433, nor equipment that satisfies the criteria of 10 CFR 50.36(d)(2)(ii) for establishment of an LCO.

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The licensee stated that the Reactor Building crane provides lifting services for the reactor and refueling area. The Reactor Building crane is not used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The NRC staff reviewed the licensee's explanation and agrees with the licensee that Criterion 1 does not apply to the Reactor Building crane.

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The licensee stated that the Reactor Building crane operability and SRs do not affect a process variable, design feature, or operating restriction that is an initial condition of DBAs or transients described in the VY's Updated Final Safety Analysis Report (UFSAR). The crane operability and SRs are related to handling and movement of a spent fuel cask and ensuing that the Reactor Building crane is inspected and tested prior to use. Additionally, the existing TS requires mechanical rail stops to be installed to prohibit movement of the cask over irradiated fuel. The TS is consistent with commitments made in response to NUREG-0612 to ensure that all cask handling operations are bounded by the DBAs and transients described in the UFSAR.

In a letter dated October 31, 2008, the licensee further explained the basis for concluding the Reactor Building crane does not meet criterion 2 by stating the following:

> Accidents that can result in the release of radioactive materials to the containment, when the drywell is open, are documented in the Vermont Yankee UFSAR. The UFSAR states that greatest potential for release occurs when the drywell head and reactor head have been removed. The UFSAR states that the only accident that could result in the release of significant quantities of fission products is the accidental dropping of a fuel bundle onto the top of the core.
>
> The design basis Fuel Handling Accident (FHA) is described in UFSAR Section 14.6.4. The FHA assumes an irradiated fuel assembly is dropped onto the reactor core from the maximum height allowed by the fuel handling equipment. The analysis assumes that the entire amount of potential energy is available for

application to the fuel assemblies involved in the accident. Also, none of the energy associated with the dropped fuel assembly is absorbed by the fuel material. The FHA is assumed to bound other credible fuel handling accidents.

The fuel handling equipment used to transport irradiated fuel from the fuel pool to the reactor core is the refueling platform and associated components (e.g., mast, grapple, controls). The refueling platform is independent of the Reactor Building crane. The Reactor Building crane is not used to move irradiated fuel from the fuel pool to the reactor core.

The single-failure-proof-crane, the procured special lifting devices, and the sling selection procedures provide reasonable assurance that the handling system used for heavy load movement near the spent fuel pool will have designed-in features to prevent a load drop. As outlined in NUREG-0612, the NRC staff accepts that provision of a single-failure-proof handling system, in conjunction with other actions implemented at VY, provides defense-in-depth against drops of loads heavier than one fuel assembly and its associated handling tool. Thus, actions and events necessary to result in a heavy load drop from the Reactor Building crane over spent fuel are not sufficiently credible that this event was included among design basis events.

The NRC staff reviewed the licensee's explanation of why Criterion 2 does not apply to the Reactor Building crane. The staff agrees with the licensee that Criterion 2 does not apply because VY uses the refueling platform for transporting irradiated fuel, which is independent of the Reactor Building crane.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The licensee stated that the Reactor Building crane is not a part of the primary success path and does not function or actuate to mitigate DBAs or transients described in the VY SAR. The Reactor Building crane is used for lifting of objects within the Reactor Building. The Reactor Building crane is not used to mitigate the consequences of any DBA or transient.

The NRC staff reviewed the licensee's explanation of why Criterion 3 does not apply. The staff agrees with the licensee that the Reactor Building crane does not function or actuate to mitigate DBAs or transients.

Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The licensee states that the Reactor Building crane is not modeled in the VY probabilistic risk assessment due to its low significance to public health and safety.

The NRC staff reviewed the VY probabilistic risk assessment (PRA). The staff agrees that the Reactor Building crane is not modeled in the VY PRA, and therefore agrees with the licensee that Criterion 4 does not apply.

The licensee further stated that the combination of hardware improvements and crane safety design features, commitments to establish safe load paths and the other commitments discussed above provide a level of assurance such that TS level administrative controls are not necessary to assure safety. Following implementation of the proposed change, the VY TRM will contain all of the requirements and information related to the Reactor Building crane that had previously been contained in the VY TS. The TRM is maintained in accordance with VY administrative processes and changes to the TRM are evaluated per the requirements of 10 CFR 50.59. These controls are adequate to ensure the Reactor Building crane is operable and capable of performing its intended functions.

NRC staff has reviewed the proposed license amendment and found that the proposed changes do not change any existing requirements and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. Therefore, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed amendment. Therefore, the NRC staff found that the proposed amendment to delete TS 3.12.G and SR 4.12.G and relocate the requirements to the VY TRM to be 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. The State official had no comments.

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 and changes SRs. 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 (73 FR 68454). 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.

Principal Contributor: Ogbonna Hopkins

Date: July 13, 2009

July 13, 2009

Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
P.O. Box 250
Governor Hunt Road
Vernon, VT 05354

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: RELOCATION OF REACTOR BUILDING CRANE TECHNICAL SPECIFICATION (TAC NO. MD9725)

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Sincerely,

/RA/

James Kim, Project Manager
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

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2. Safety Evaluation

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Accession No.: ML091740040 *See memo dated June 12, 2009

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