Entergy

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October 31, 2008
BVY 08-079

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: (a) Letter, VYNPS to USNRC, "Technical Specification Proposed Change No. 280 Relocation of Reactor Building Crane Technical Specifications," BVY 08-059, dated September 22, 2008

Subject: **Vermont Yankee Nuclear Power Station**
Docket No. 50-271, License No. DPR-28
Technical Specifications Proposed Change No. 280, Supplement 1
Relocation of Reactor Building Crane Technical Specifications

Dear Sir or Madam,

In Reference (a), Entergy Nuclear Operations, Inc. (Entergy) proposed to amend Operating License DPR-28 for Vermont Yankee Nuclear Power Station (VY) to relocate Technical Specification requirements to the Technical Requirements Manual. During the acceptance review, the NRC staff requested additional information related to how the change satisfied the requirements of 10CFR50.36(d)(2)(ii)(B) Criterion 2 and on the proposed administrative controls. Attachment 1 provides the requested information.

This supplement to the original license amendment request does not change the scope or conclusions in the original application, nor does it change the Entergy's determination of no significant hazards consideration

There are no new regulatory commitments made in this letter.

If you have any questions on this transmittal, please contact Mr. David Mannai at (802) 452-3304.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on October 31, 2008.

Sincerely,

Michael J. Colomb
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment
cc listing (next page)

A001
NRR

cc: Mr. Samuel J. Collins (w/o attachments)
Regional Administrator, Region 1
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Mr. James S. Kim, Project Manager
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Mr. David O'Brien, Commissioner
VT Department of Public Service
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Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Change 280
Supplement 1

Response to Request for Additional Information

Vermont Yankee Nuclear Power Station
Proposed Change 280, Supplement 1
Response to Request for Additional Information

Requested Information:

Provide additional justification for how the proposed change satisfies the requirements of 10CFR50.36(d)(2)(ii)(B) Criterion 2 and why the proposed administrative controls are adequate.

Response:

10CFR50.36(a)(2)(ii)(B) Criterion 2, requires Technical Specifications (TS) for "A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

Accidents that can result in the release of radioactive materials to the containment, when the drywell is open, are documented in the Vermont Yankee Updated Final Safety Analysis (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 TS related to reactor building crane operability (TS 3.12.G and TS 4.12.G) were established by License Amendment No. 29 to Facility Operating License No. DPR-28 (Reference 1). License Amendment No. 29 established requirements to ensure that the reactor building crane would be inspected and tested before use to lift a spent fuel cask and established requirements that would prohibit the movement of a spent fuel cask over irradiated fuel assemblies within the spent fuel pool.

Entergy has also made commitments related to the control of heavy loads. The reactor building crane was identified as within the scope of NUREG-0612 and a number of improvements were implemented. For example, 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.

In addition to the hardware improvements implemented to satisfy commitments made to NUREG 0612 a number of administrative improvements were committed to. Examples include:

- Maintenance procedures were revised to define safe load paths for major loads.
- Procedures were revised to include training and qualification requirements for crane operators, sling selection criteria, crane inspections prior to use and supervisory oversight of heavy lift operations.
- Special lifting devices were identified and obtained and periodic non-destructive examination was scheduled to monitor the condition of lifting devices.

Entergy's position is 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.

Entergy's proposal to relocate the current crane operability and surveillance requirements to the Technical Requirements Manual (TRM) is acceptable for the following reasons:

- Commitments made in response to NUREG 0612 are managed in accordance with NEI 99-04 "Guidelines for Managing NRC Commitment Changes", Revision 0, dated July 1999. This ensures a safety review of commitment changes is performed and the NRC notified of changes to regulatory commitments as required.
- Additionally, the TRM is "included by reference" in the UFSAR. As such changes to the TRM are evaluated in accordance with 10CFR50.59. This provides requirements that future changes to the operability and surveillance requirements will be evaluated to ensure that they do not create the possibility of an accident that is different than or that has greater consequences that the design basis FHA without prior NRC approval. Changes made under 10CFR50.59 are reported to the NRC as required.
- Finally, the industry and NRC have worked to establish the expected scope and content of the TS to ensure compliance with 10CFR 50.36 requirements. For a General Electric Boiling Water Reactor (BWR), NUREG-1433 Revision 3, establishes the current expectations. Reactor Building Crane operability and surveillance requirements are not within the scope of NUREG-1433.

Reference:

1. Letter, USNRC to YAEC, dated January 28, 1977