Entergy

Entergy Nuclear Operations, Inc
Vermont Yankee
P.O. Box 0250
320 Governor Hunt Road
Vernon, VT 05354
Tel 802 257 7711

January 21, 2009
BVY 09-001

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

Reference: Letter USNRC to VYNPC, "TMI Action Plan Item II.K.3.3, Reporting of Relief Valve Failures and Challenges", NVY 82-44, dated March 30, 1982

Subject: **Vermont Yankee Nuclear Power Station**
Docket No. 50-271 (License No. DPR-28)
Cycle 26 10CFR50.59 Report

Dear Sir or Madam:

In accordance with 10CFR50.59, attached is the Vermont Yankee Cycle 26 10CFR50.59 Report. This report contains a summary of the 50.59 Evaluation that was performed between June 7, 2007 and November 10, 2008.

Additionally, in accordance with the referenced letter, Vermont Yankee reports that there were no Main Steam Relief Valve or Safety Valve failures or challenges during this period.

There are no new commitments contained in this submittal.

If you have any questions or require additional information, please contact Mr. David Mannai at (802) 451-3304.

Sincerely,

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

Attachment: Vermont Yankee Cycle 26 10CFR50.59 Report
cc: (next page)

IE47
NRR

cc: Mr. Samuel J. Collins
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. James S. Kim, Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O8C2A
Washington, DC 20555

USNRC Resident Inspector
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

BVY 09-001
Docket No. 50-271

<u>Attachment</u>

Vermont Yankee Nuclear Power Station

Vermont Yankee Cycle 26 10CFR50.59 Report

Vermont Yankee Cycle 26 10CFR50.59 Report

Between June 7, 2007 and November 10, 2008, Vermont Yankee (VY) implemented one change requiring evaluation in accordance with 10CFR50.59. This report provides a summary of the evaluation performed for this change. The evaluation was reviewed by the On-Site Safety Review Committee (OSRC), approved by the OSRC Chairman and concluded that prior Nuclear Regulatory Commission review and approval was not required.

10CFR50.59 Evaluation No. 2008-01 Rev. 0, Engineering Change (EC) 1604, "Independent Spent Fuel Storage Facility"

VY established an interim Independent Spent Fuel Storage Installation (ISFSI) licensed under the provisions of 10CFR72 Subpart K "General License for the Storage of Spent Fuel at Power Reactor Sites." Engineering Change (EC) 1604 documented and evaluated the integrated impact of facility changes and operational considerations (e.g., required plant modifications, heavy loads issues, impact on Structures, Systems and Components (SSCs) Important to Safety (ITS)) that are required to design, construct and implement dry fuel storage at VY. EC 1604 provides the necessary analysis to load the first 5 HI-STORM storage casks. The VY ISFSI was designed, evaluated and licensed as required by 10CFR72, Amendment 2 of the HOLTEC International Certificate of Compliance (CoC) for Spent Fuel Storage Casks (Certificate no. 1014) and the HOLTEC International Final Safety Analysis Report for the HI-STORM 100 Cask system, Revision 4 (CFSAR).

Accidents previously evaluated in the UFSAR include the Abnormal Operational Transients and Design Basis Accidents identified in UFSAR Chapter 14, a Station Blackout (SBO) event, 10CFR50 Appendix R fire events and an Anticipated Transient without Scram (ATWS) event. The changes that support implementation of ISFSI include installing an ISFSI pad, making a number of plant modifications to the haul path to support the equipment loading requirements, performing evaluations of the load path and lay-down areas, the use of ancillary equipment like welding machines, vacuum drying equipment and helium backfilling equipment and the actual loading and cask movement activities. Other temporary tools and equipment are used for monitoring dose and for personnel safety reasons. Based on a review of the ISFSI activities, it is concluded that they do not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

Malfunctions of ITS SSCs previously evaluated in the UFSAR include postulated failures of SSCs to perform their intended design functions. To affect the likelihood of an SSC malfunction, an ISFSI activity would need to have a direct or indirect affect on an ITS SSC described in the UFSAR. The majority of modifications made to support ISFSI involve passive structures (e.g., ISFSI pad, grounding system, drainage system, haul path strengthening) and have no direct or indirect impact on equipment important to safety. The other modifications (e.g., ISFSI Temperature monitoring, Reactor Building Crane Modifications are installed to satisfy ISFSI Technical Specifications and to accommodate lifting fuel assemblies into the Multi Purpose

Canister (MPC). These changes also do not have a direct or indirect impact on ITS SSCs. Based on a review of ISFSI activities, it is concluded that they do not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

Accidents with dose consequences previously evaluated in the UFSAR include the Design Basis Accidents evaluated in UFSAR Chapter 14 or malfunctions evaluated in the UFSAR. ISFSI related activities do not have the potential to impact systems that mitigate the consequences of the design basis accidents since heavy load paths and lay-down areas are evaluated and a drop of the loaded cask is not considered credible based on use of the single failure proof crane. Additionally, Primary and Secondary Containment systems are required to be operable during cask movement. Based on a review of the ISFSI activities, it is concluded that they do not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR nor do they result in any increase in the consequences of a malfunction of a structure, system or component important to safety previously evaluated in the UFSAR.

The accident types previously evaluated in the UFSAR include; high energy pipe breaks that challenge the primary containment boundary, high energy pipe breaks outside primary containment that challenge containment isolation and equipment qualification, refueling accidents, control rod drop events, fires that challenge safe shutdown, Station Blackout, and ATWS events. Use of the HOLTEC 100 System involves construction of an ISFSI pad, a number of other supporting modifications to the facility and moving fuel from the fuel pool to the pad using an MPC, HI-TRAC transfer cask and a HI-STORM storage cask. Support evolutions like welding and helium backfilling will be performed using approved procedures that have been subject to review under 10CFR50.59 and performed under existing work control procedures (e.g., Fire Permits, Radiation Work Permits). The storage of loaded canisters in their concrete and steel overpacks on the ISFSI pad is a totally passive system relying only on natural thermal convection for cooling. The CFSAR has evaluated a number of off-normal and accident conditions and a review of site specific considerations did not reveal any credible accident of a different type than those previously evaluated in the UFSAR. Based on this, the installation and operation of the ISFSI on the VY site does not create the possibility of a different type of accident than any previously evaluated in the UFSAR.

Activities that have the potential to impact ITS SSCs involve the handling of heavy loads within the Reactor Building and along the haul path. Analysis has been perform on the load paths, lay down areas and the haul path to ensure loads can be accommodated by existing structures and that ITS SSCs are not impacted. In some cases modification or temporary reinforcement was required to provide this assurance. The heavy loads will be performed using the Refuel Bridge (for fuel moves) the Reactor Building crane for MPC, HI-STORM and HI-TRAC lid lifts and lifts of other ancillary equipment. Load paths and lay-down areas have been identified and evaluated consistent with VY commitments to NUREG 0612. The Reactor Building crane is a single failure proof crane and a malfunction of the crane, which would result in a cask drop accident, has been determined to not be a credible event. The storage of loaded canisters in their concrete and steel overpacks on the ISFSI pad is a totally passive system relying only on natural thermal convection for cooling. There is no credible equipment malfunction that would result in a release of radioactivity from the stored fuel. Therefore, based on a review of ISFSI activities, it is concluded that they will not create a possibility for a malfunction of an ITS SSC with a different result than any previously evaluated in the UFSAR.

Activities associated with implementation of ISFSI do not have direct or indirect impact on design basis limits for fission product barriers associated with the Reactor Coolant System Pressure Boundary or Peak Containment Pressure. Since the activities deal with movement of irradiated fuel, the design basis limit for Peak Clad Temperature (PCT) needs to be considered. The UFSAR specifies a PCT limit of 2200 degrees F. Also, the cask containment system will serve as a second fission product barrier. This is not specified in the UFSAR. The fuel cladding of the fuel assemblies to be loaded constitutes the first fission product barrier. The license process for the HI-STORM system established the specific parameters for fuel selection in terms of cladding temperature, criticality control, protection of cladding from degrading environments, and physical protection from external events. These requirements are contained in Appendix A, Technical Specifications, and Appendix B, Approved Contents and Design Features, to the HOLTEC Certificate of Compliance (CoC) Docket No. 1014, Amendment 2. The maximum peak clad temperature that results from an event associated with the cask is maintained below 1058 degrees F for accident and off-normal events. Therefore, based on a review of ISFSI activities, it is concluded that they do not impact any UFSAR identified design basis limits for fission product barriers.

The UFSAR does not discuss the methods of evaluation associated with the design, loading, and storage of the HOLTEC dry cask system. The USNRC has reviewed the analytical methods used by HOLTEC as part of the licensing process under 10CFR72. The Safety Evaluation Report for Amendment 2 of the HOLTEC CoC concludes by stating that: "The staff determined that, unless otherwise noted in this SER, all analytical methods used by the applicant, that provide the basis for design modifications and the addition to the list of approved cask contents for the HI-STORM 100 Cask System proposed in Amendment 2, are acceptable." Also, methods used for performance of the load drop analysis are consistent with VY commitments to NUREG 0612. Consequently, the design and evaluation of the Holtec Dry Cask Storage system does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.