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

February 12, 2015

Vice President, Operations
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 TO RENEWED FACILITY OPERATING LICENSE RE: ELIMINATE OPERABILITY REQUIREMENTS FOR SECONDARY CONTAINMENT WHEN HANDLING SUFFICIENTLY DECAYED IRRADIATED FUEL OR A FUEL CASK (TAC NO. MF3068)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 262 to Renewed Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 14, 2013, as supplemented by letters dated June 9, 2014, August 6, 2014, and October 9, 2014.

The amendment revises the technical specifications to eliminate the operability requirements for secondary containment when handling sufficiently decayed irradiated fuel or a fuel cask following a minimum of 13 days after the permanent cessation of reactor operation.

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

Sincerely,

James Kim

James Kim, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures:
1. Amendment No. 262 to 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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 262
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 November 14, 2013, as supplemented by letters dated June 9, 2014, August 6, 2014, and October 9, 2014, 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 Renewed 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. 262, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

3. The license amendment becomes effective 13 days after the licensee's submittal of the certifications, as required by 10 CFR 50.82(a)(1)(i) and (ii) and shall be implemented within 60 days from the amendment's effective date.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing Branch IV-2 and
Decommissioning Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. DPR-28 and Technical Specifications

Date of Issuance: February 12, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 262

RENEWED FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following page of the Renewed 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 contains marginal lines indicating the areas of change.

Remove	Insert
51	51
155	155
155a	155a
156	156
157	157

D. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any Byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components.

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 renewed 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. 262 are 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:

1. This paragraph deleted by Amendment No. 206, October 22, 2001.

2. This paragraph deleted by Amendment 131, 10/07/91.

Table 3.2.3 (page 1 of 1)
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation

TRIP FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	ACTIONS WHEN REQUIRED CHANNELS ARE INOPERABLE	TRIP SETTING
1. Low Reactor Vessel Water Level	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b)]	2	Note 1	≥ 127.0 inches
2. High Drywell Pressure	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a)]	2	Note 1	≤ 2.5 psig
3. High Reactor Building Ventilation Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b), (c), (d)]	1	Note 1	≤ 14 mR/hr
4. High Refueling Floor Zone Radiation	RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel[(a), (b), (c), (d)]	1	Note 1	≤ 100 mR/hr

(a) With reactor coolant temperature > 212 °F.

(b) During operations with potential for draining the reactor vessel.

(c) During movement of recently irradiated fuel assemblies in secondary containment.

(d) During Alteration of the Reactor Core.

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS

shutdown condition, the actions and completion times of Specification 3.7.B.4.b shall apply. After seven days with an inoperable train of the Standby Gas Treatment System during refueling or cold shutdown conditions requiring secondary containment integrity, the operable train of the Standby Gas Treatment System shall be placed in operation and its associated diesel generator shall be operable, or the actions and completion times of Specification 3.7.B.4.b shall apply.

4. With two trains of the Standby Gas Treatment System inoperable, or as made applicable by Specification 3.7.B.3:

   a. With the reactor in the run mode, startup mode, or hot shutdown condition, the reactor shall be placed in hot shutdown within 12 hours and cold shutdown within 36 hours.

   b. During movement of recently irradiated fuel assemblies in the secondary containment, during core alterations, or during operations with the potential for draining the reactor vessel, immediately:

3.7 LIMITING CONDITIONS FOR OPERATION

i. Suspend movement of recently irradiated fuel assemblies in secondary containment; and

ii. Suspend core alterations; and

iii. Initiate action to suspend operations with the potential for draining the reactor vessel.

C. Secondary Containment System

1. Secondary Containment Integrity shall be maintained during the following modes or conditions:

a. Whenever the reactor is in the Run Mode, Startup Mode, or Hot Shutdown condition*; or

4.7 SURVEILLANCE REQUIREMENTS

C. Secondary Containment System

1. Secondary containment capability to maintain a 0.15 inch of water vacuum under calm wind ($2<\bar{u}<5$ mph) conditions with a filter train flow rate of not more than 1,550 cfm, shall be demonstrated at least quarterly.

* NOTE: The reactor mode switch may be changed to either the Run or Startup/Hot Standby position, and operation not considered to be in the Run Mode or Startup Mode, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided:

1. Reactor coolant temperature is $\leq$ 212°F;
2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
3. No core alterations are in progress.

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
b. During movement of recently irradiated fuel assemblies in secondary containment; or	
c. During alteration of the Reactor Core; or	
d. During operations with the potential for draining the reactor vessel.	

3.7 LIMITING CONDITIONS FOR OPERATION

2. With Secondary Containment Integrity not maintained with the reactor in the Run Mode, Startup Mode, or Hot Shutdown condition, restore Secondary Containment Integrity within four (4) hours.

3. If Specification 3.7.C.2 cannot be met, place the reactor in the Hot Shutdown condition within 12 hours and in the Cold Shutdown condition within the following 24 hours.

4. With Secondary Containment Integrity not maintained during movement of recently irradiated fuel assemblies in secondary containment, during alteration of the Reactor Core, or during operations with the potential for draining the reactor vessel, immediately perform the following actions:

   a. Suspend movement of recently irradiated fuel assemblies in secondary containment; and

   b. Suspend alteration of the Reactor Core; and

   c. Initiate action to suspend operations with the potential for draining the reactor vessel.

4.7 SURVEILLANCE REQUIREMENTS

2. Intentionally blank.

3. Intentionally blank.

4. Intentionally blank.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 262

TO RENEWED 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 November 14, 2013 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML13323A516), as supplemented by letters dated June 9, 2014 (ADAMS Accession No. ML14163A008), August 6, 2014 (ADAMS Accession No. ML14224A012), and October 9, 2014 (ADAMS Accession No. ML14288A317), Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy, the licensee), submitted a license amendment for Vermont Yankee (VY). The amendment request eliminates operability requirements for secondary containment when handling sufficiently decayed irradiated fuel or a fuel cask using Technical Specification (TS) Task Force (TSTF) Change Traveler 51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," Revision 2.

The supplemental letters dated June 9, 2014, August 6, 2014, and October 9, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration determination, as published in the *Federal Register* on September 16, 2014 (79 FR 55511).

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include the TSs as part of the license. The U.S. Nuclear Regulatory Commission's (NRC's) regulatory requirements related to the content of TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, 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; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in the TSs.

The following four criteria, as specified in 10 CFR 50.36(c)(2)(ii), determine whether particular items are required to be included in the TS LCOs:

(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 design basis accident 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 DBA and transient analyses, as discussed in Criteria 2 and 3, include any design basis events described in the safety analysis report, not just those events described in the accident analysis chapter.

In addition, the following regulatory requirements and guidance were considered by the NRC staff in its review of the amendment:

Section 50.67 of 10 CFR, "Accident source term," states that:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE);
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE);
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000, provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. The RG 1.183 provides guidance to licensees on the acceptable application of the alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition – Transient and Accident Analysis," Chapter 15, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000, provides guidance to the staff for the review of alternative source term amendment requests. SRP 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183.

License Amendment No. 223, dated March 29, 2005 (ADAMS Accession No. ML041280490), "Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: Alternative Source Term (TAC No. MC0253)," used an AST methodology for analyzing the radiological consequences of four DBAs using RG 1.183. The fuel handling accident (FHA) was one of the DBAs that was analyzed.

The NRC staff based its approval of the licensee's request on the reference values in 10 CFR 50.67, the accident specific guideline values in Regulatory Position 4.4 of RG 1.183 and Table 1 of SRP Section 15.0.1.

3.0 TECHNICAL EVALUATION

3.1 Description of Changes

The proposed changes would revise select TSs associated with the handling of irradiated fuel or a fuel cask. Specifically, the changes would eliminate operability requirements for secondary containment (TS 3.7.C), the Standby Gas Treatment (SGT) system (TS 3.7.B.4), and the reactor building ventilation isolation and the SGT system initiation instrumentation (TS Table 3.2.3) when handling irradiated fuel or a fuel cask. The proposed changes would require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined by Entergy to be 13 days and will be defined in the TS Bases. The staff's assessment of the period of sufficient radioactive decay is addressed in Section 3.3.2.2 of this safety evaluation.

The licensee developed a revised FHA (or refueling accident) to provide a basis for the proposed changes. The licensee stated that it used the AST methodology described in Regulatory Guide 1.183 for the revised FHA.

3.2 Evaluation for TSTF-51

According to Entergy's November 14, 2013, application, as supplemented by letter dated June 9, 2014, the proposed changes are based on the generic changes including those in the "Technical Specification Task Force Improved Standard Technical Specifications Change Traveler: *Revise* *containment requirements during handling irradiated fuel and core alterations,"* TSTF-51-A, Rev. 2 (ADAMS Accession No. ML040400343). The original TSTF-51-A, Rev. 2 was approved on November 1, 1999 (ADAMS Accession No. ML993190284).

TSTF-51, in part, states:

> The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by *analysis* [emphasis added] that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10 CFR 100 [Reactor Site Criteria]) [or 10 CFR 50.67].

Following a reactor shutdown, the decay of the short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed TS changes take advantage of a specific decay period to reduce the radionuclide inventory available for release in the event of an FHA. Entergy calculated this specific decay period to be 13 days. Following the 13-day decay period, the primary success path for mitigating the FHA no longer includes the operability of the subject engineered safety features (ESF) components, as discussed in Section 3.1 of this safety evaluation (SE). Fuel that has not decayed for 13 days or longer is termed "recently irradiated fuel" and the ESF features, discussed above, must remain operable by the VY technical specifications, when moving such fuel.

On November 7, 2013, the NRC sent a letter to the TSTF (ADAMS Accession No. ML13246A358). This letter and subsequent meetings with the TSTF identified potential issues with the plant specific adoption of several TSTFs, including TSTF-51. One potential issue (Issue c in the November 7, 2013, letter) relates to the analyses needed to justify proposed changes using TSTF-51. The NRC staff stated that they may require an additional analysis to show that the dose consequences are less limiting than the current FHA.

By letter dated June 9, 2014, Entergy responded to the NRC staff's request for additional information (RAI) (ADAMS Accession No. ML14163A008), which requested a load drop analysis of those loads authorized to be moved over the stored spent fuel assemblies. In the response, Entergy proposed three changes to the original submittal. Entergy first proposed to retract its request to remove references to "Core Alterations" from the VY TSs. Secondly, Entergy proposed that the changes to the TSs not be implemented while there is fuel in the VY reactor vessel. The requested approval date of the proposed TSs would be contingent upon the docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii). Lastly, the approval date of the proposed TSs is also contingent upon approval of the proposed changes being a minimum of 13 days after the permanent cessation of operations.

The NRC staff evaluated Entergy's June 9, 2014, response to the NRC staff's request for additional information. The changes to the original submittal were evaluated against the assumptions in the design basis FHA. Since the proposed changes will not be in effect until the reactor is permanently defueled, the FHA in the reactor cavity is not impacted. Once the fuel is in the spent fuel pool and there has been at least 13 days since operations (13 days of fuel decay), the proposed TS changes will be in effect.

The NRC staff performed an analysis, as described in Section 3.3 of this SE, to confirm that the licensee's analyses of the radiological consequences are within the acceptance criteria of 10 CFR 50.67, RG 1.183, and SRP 15.0.1 for an FHA after 13 days of fuel decay. Since Entergy did not provide a limiting FHA without certain ESF systems operable (See Section 3.1 of this SE) for fuel decayed less than 13 days (i.e., for fuel that has been "recently irradiated"), the staff's acceptance of the proposed changes is, therefore, contingent upon the implementation date of the proposed changes being after: (1) the docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii), and (2) a minimum of 13 days after the permanent cessation of operations.

As part of its evaluation of adding the term "recently" to the TSs, the licensee proposed two commitments to address the TSTF-51 reviewer's notes. The NRC staff requested additional information regarding these two commitments. Details for the RAI response and its review are presented below.

In the August 6, 2014, response to an RAI, the licensee stated that the assessment of the ventilation system and radiation monitor availability will be completed prior to the use of the license amendment, once approved. In other words, the licensee is committing to perform such an assessment prior to any fuel handling operations with openings in the secondary containment following permanent defueling of the VY reactor, which occurred in January 2015. The NRC staff has reviewed the response and found it to be acceptable because the licensee has confirmed that ventilation and radiation monitoring will be available. In addition, the staff agrees that the licensee has sufficient time to complete the assessment.

Similarly, the licensee stated that the regulatory commitment to establish contingency methods to ensure prompt closure of openings in the secondary containment will be completed prior to the implementation of the license amendment. In the licensee's response of August 6, 2014, it states:

> Entergy notes that there will be inherent variability in each contingency plan that is implemented prior to a period of fuel handling operations due to the unique set of conditions likely to be present in terms of the number, and location of open penetrations in the secondary containment.

The NRC staff has reviewed the response and found it to be acceptable because the licensee provided the contingency plans for prompt closure of openings in the proposed license amendment and the licensee has sufficient time to develop and provide training on robust measures to ensure that the contingency methods are implemented promptly.

3.3 Accident Dose

3.3.1 Alternative Source Term

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, that provided a mechanism for licensed power reactors to voluntarily replace the traditional accident source term used in their DBA analyses with ASTs. The regulatory guidance for the implementation of these ASTs is provided in RG 1.183. Under 10 CFR 50.67, a licensee seeking to use the AST is required to apply for a license amendment and the application is required to contain an evaluation of the consequences of the DBAs.

The NRC has already approved the use of an AST at VY by License Amendment No. 223. In License Amendment No. 223, the FHA assumes a fuel decay period of 24 hours, credit for the reactor building being closed, and collection of fission products by the SGT system to the main stack.

In the proposed license amendment, Entergy provided a revised assessment of the FHA to support the proposed changes to the TSs. The revised assessment assumes an FHA without credit for secondary containment, control room high efficiency air filtration or the SGT system, and assumes the fuel has been decayed for 13 days before fuel handling occurs.

Entergy stated, and the NRC staff concurs, that the FHA is the limiting event with regards to the proposed TS changes. Section 3.3.2 of this SE, provides the results of the staff's review of the licensee's analyses. Although the staff did confirmatory analyses, the staff's approval of the requested changes was based on the information docketed by the licensee and on the staff's finding that the reviewed methods, inputs, and assumptions used in the licensee's analyses are acceptable.

3.3.2 Fuel Handling Accident Radiological Consequences

3.3.2.1 Licensee's Assessment

The FHA analysis postulates that a spent fuel assembly is dropped during movement of the fuel 13 days after shutdown. The kinetic energy developed in this drop is conservatively assumed to be dissipated by the damage to the fuel cladding.

The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously from the fuel to the overlying fuel pool to the secondary containment building. This radioactivity is released to the environment from the secondary containment over a 2-hour period. Consistent with the proposed change, Entergy does not credit the secondary containment, the SGT system, the reactor building ventilation isolation instrumentation or the SGT system initiation instrumentation. Therefore, Entergy assumes that the release to the environment is an unfiltered ground-level release. The release is assumed to be via the reactor building blowout panels.

Fission products released from the damaged fuel are decontaminated by passage through the pool water, with the degree of decontamination, depending on their physical and chemical form. Entergy assumed no decontamination for noble gases, a factor of 200 decontamination of radioiodines, and retention of all aerosol and particulate fission products. Entergy evaluated the maximum 2-hour TEDE to an individual located at the exclusion area boundary and the 30-day TEDE to an individual at the outer boundary of the low population zone. The resulting doses were found to be less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria and are less than the 10 CFR 50.67 criteria.

Entergy evaluated the dose to operators in the control room. It was assumed that the control room would not be isolated during the event. The control room ventilation system draws in

3700 cubic feet per minute (cfm) of unfiltered outside air. Entergy analyzed the control room dose over a 30-day period. The resulting 30-day TEDE to an individual in the control room was found to be less than the 10 CFR 50.67 criteria.

3.3.2.2 NRC Staff's Assessment

The NRC staff evaluated Entergy's proposed changes to the current licensing basis FHA. The staff performed a confirmatory analysis assuming 13 days of decay time before moving fuel without certain safety systems operable (i.e., no credit for the secondary containment, the SGT system, the reactor building ventilation isolation instrumentation or the SGT system initiation instrumentation).

The NRC staff found that the decay time assumed by Entergy is consistent with RG 1.183, Regulatory Position 3.1. Regulatory Position 3.1 states that for events postulated to occur while the facility is shutdown, radioactive decay from the time of shutdown may be modeled.

The change, which no longer credits several safety systems after 13 days of fuel decay, was modeled by Entergy, consistent with RG 1.183, Appendix B, and Regulatory Position 5.3. Entergy's FHA with an open containment model does not credit the secondary containment, SGT system, or the reactor building ventilation. Consistent with Regulatory Position 5.3, the radioactivity that escapes the fuel pool is released to the environment over a 2-hour time period.

In Entergy's response dated October 9, 2014 (ADAMS Accession No. ML14288A317), to the NRC staff's RAI, Entergy compared the decontamination factors and number of damaged assemblies for an FHA in the spent fuel pool to those for an FHA over the reactor core. Entergy stated that the minimum amount of water above a postulated dropped and damaged assembly lying on top of a spent fuel rack is 20.67 feet, rather than the 23 feet assumed for the FHA over the reactor core. Entergy also stated that although the decontamination factor in the spent fuel pool is reduced from a factor of 200 to 125 (due to there being 20.67 feet of water above the postulated damaged fuel assembly), this reduction is more than offset by the difference in the number of postulated damaged fuel rods in the spent fuel pool.

RG 1.183 states that if the depth of water above the damaged fuel is 23 feet or greater, the overall effective decontamination factor is 200. If the depth of the water above the top of the fuel is less than 23 feet, the decontamination factor will be determined on a case-by-case basis using a model in the report by G. Burley entitled, "Evaluation of Fission Product Release and Transport." The NRC staff assessed the calculation performed by Entergy using the Burley model and confirmed the calculated decontamination factor of 125. Based upon this decontamination factor and the estimated number of fuel rods damaged for an FHA over the spent fuel pool, the staff agrees that the release from the water for an FHA over the spent fuel pool will be bounded by the release over the reactor core.

In Attachment 1 of the RAI dated October 9, 2014, the licensee stated that the postulated number of damaged fuel rods, resulting from an FHA in the spent fuel pool, would be much lower than the postulated number of damaged fuel rods for an FHA in the reactor vessel. The licensee determined that the number of damaged fuel rods, resulting from an FHA, would decrease from 193 to 98, as a result of the different fuel handling configuration in the spent fuel pool, relative to the reactor vessel. The licensee based this conclusion on the fuel damage methodology, as described in the GESTARII licensing topical report (General Electric Standard Application for Reactor Fuel, June 2000), which had also been used to determine the fuel damage in the analysis of record for the fuel handling accident in the reactor vessel.

The licensee stated that the final operating core for VY contains 14 GE14 fuel assemblies and 354 Global Nuclear Fuel (GNF)2 fuel assemblies, and described the characteristics of the assemblies as follows:

- Both the GE14 and GNF2 fuel assembly types are 10 x 10 arrays that contain 92 fuel rods.
- Of the 92 fuel rods, 78 are full length rods and 14 are part length rods.
- The rod cladding for GNF2 fuel assemblies is thinner than it is for GE14 fuel assemblies, which results in a lower fuel rod compression failure strength for the GNF2 fuel assembly rods.
- The GNF2 fuel assemblies are heavier than the GE14 fuel assemblies.

The licensee also described the following assumptions used in the determination of fuel rod damage:

- A GNF2 fuel assembly is dropped directly above another GNF2 assembly seated in the fuel storage racks.
- All 92 rods in the dropped assembly are assumed to fail.
- If the analysis results project fewer than 78 damaged rods in the impacted assembly, all of the damaged rods in the impacted assembly are assumed to be full length rods to maximize the release.

The NRC staff evaluated the assumed accident configuration for the FHA. In going from the reactor to the spent fuel pool, the maximum potential energy of the postulated dropped assembly substantially decreases because the maximum potential lift height above fuel decreases from 34 feet to less than 3 feet. If the fuel was assumed to drop in a location with a longer vertical drop (e.g., into an open storage slot or outside the racked area in the spent fuel pool), the spent fuel storage rack structure would offer protection for nearby assemblies. Similarly, the rack structure extends above the top of the fuel assemblies seated in the rack, which would prevent direct impact on more than one assembly and any secondary impact if the dropped assembly were to fall across the top of the racks. Also, because the GNF2 fuel assemblies have lower compressive strength and higher weight than the other assembly type in the current core, a drop of a GNF2 assembly on another GNF2 assembly would result in a greater postulated fuel pin damage than other combinations of fuel assembly types. Therefore, the staff agrees that the assumed accident configuration is limiting for an FHA in the spent fuel pool.

The NRC staff also evaluated the postulated number of damaged pins for the above accident configuration, considering the methodology described in the GESTAR II topical report. The staff determined that the number of postulated damaged pins (98) is consistent with the methodology described in the topical report and represents a conservative estimate of fuel damage for the postulated FHA in the spent fuel pool.

The NRC staff also confirmed with the licensee that the release point from the reactor building to the control room uses the most limiting combination of release and receptor locations. Since Entergy is assuming a ground level release, the staff finds it acceptable for the licensee to model the FHA using the Control Room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) atmospheric dispersion factors for an FHA with a ground level release. These were previously approved in License Amendment No. 223.

Using the new analyses assumptions described above, the NRC staff's confirmatory analyses of VY's FHA yields results for the EAB, LPZ and Control Room doses that are less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria. Because the proposed modeling changes are consistent with RG 1.183, and yield doses within the RG and SRP acceptance criteria, the NRC staff finds these changes acceptable.

3.3.3 Fuel Cask Movement

The NRC requested licensees to address control of heavy load movements in 1980. The NRC staff provided regulatory guidelines to support this action in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36." Implementation of these guidelines assures safe handling of heavy loads in areas where a load drop could have an impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Section 5.1 described acceptable approaches for the control of heavy loads, which assure either the potential for a load drop is extremely small or the potential consequences of a load drop are acceptably small.

Section 5.1.1 of NUREG-0612 provides general guidelines for reducing the likelihood of dropping heavy loads, which includes specific criteria for establishing safe load paths; procedures for load handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops. The guidelines in Section 5.1.4 address approaches for operations within the reactor building of a boiling-water reactor, such as VY. These alternatives include using a single-failure-proof crane for increased handling system reliability or performing load drop consequence analyses to assess the effects of dropped loads on plant safety and operations. Section 5.1.6 specifically addresses measures to further reduce the probability of a load handling accident through installation and operation of a highly reliable load handling system. These measures include the use of a single failure-proof crane to improve reliability through increased factors of safety and through redundancy or duality in certain active components.

During handling of the fuel cask, the existing VY TSs require the operability of the secondary containment, the SGT system, and the associated initiation instrumentation. VY TS 3.7.C.1.b requires that secondary containment integrity be maintained, in part, during movement of the fuel cask in secondary containment; and, VY TS 3.7.B.1 requires the SGT system to be operable at all times when secondary containment integrity is required. During movement of the fuel cask in secondary containment, if secondary containment is not maintained, VY TS 3.7.C.4.a requires the suspension of fuel cask movement in secondary containment. During movement of the spent fuel cask in the secondary containment, if two trains of the SGT system are inoperable or one train is inoperable for more than 7 days in the cold shutdown or refueling condition, VY TS 3.7.B.4.b requires the immediate suspension of fuel cask movement. Finally, VY TS 3.2.C requires the reactor building ventilation isolation and SGT system initiation instrumentation trip functions for the reactor building ventilation and refueling floor zone high radiation trips to be operable, in part, during movement of the fuel cask in secondary containment. The licensee proposed to revise these TSs to delete references to movement of the fuel cask in secondary containment from these specifications because a radiological release from a handling accident involving the fuel cask is not considered credible.

Section 12.2.2.2 of the VY Updated Final Safety Analysis Report (UFSAR) described that the reactor building crane is a 110-ton capacity overhead bridge crane that provides services for the reactor and refueling area. The crane handles the spent fuel shipping cask and storage cask components.

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 satisfied the criteria for dual load path or "single-failure-proof" cranes, and, with issuance of License Amendment No. 29 to the VY operating license on January 28, 1977 (ADAMS Accession No. ML011590477), 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, in response to the 1980 request by the NRC staff. 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 Report (SER) transmitted by letter dated June 27, 1984. Section 12.2.2.2 of the VY UFSAR described that the VY Control of Heavy Loads Program Document specifies the commitments made as input to the SER and how they are implemented at VY.

By letter dated July 13, 2009, the NRC issued License Amendment No. 239 to the VY Operating License (ADAMS Accession No. ML091740040). This amendment permitted relocation of technical specification limits on the operation of the reactor building crane to the VY Technical Requirements Manual, a licensee-controlled document. These limits included crane functionality requirements and requirements to install mechanical rail stops to prevent the movement of a fuel cask over irradiated fuel.

The NRC guidelines contained in NUREG-0612 served as the basis for licensing control of heavy load movements at VY. In the NRC staff SER for Amendment No. 239 to the VY Operating License, the NRC documented the following basis for excluding postulated cask drops from consideration as design-basis events:

> 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.

Since a heavy load drop from the reactor building crane was not included among design basis events, movement of the spent fuel cask is not an initial condition of a design basis event that could challenge a fission product barrier and require the operation of secondary containment systems to mitigate a release. Consistent with Criterion 2 of 10 CFR 50.36(c)(2)(ii), operability of secondary containment, the SGT system, and the high radiation trip initiation instrumentation for reactor building ventilation isolation and SGT system operation is not necessary during fuel cask movement. Therefore, deletion of the reference to fuel cask movement in the affected TSs (i.e., TS 3.7.C.1.b, TS 3.7.C.4.a, TS 3.7.B.1, TS 3.7.B.4.b, and TS 3.2.C) is acceptable.

3.4 Control Room Habitability

On June 21, 2005, Entergy submitted their response to Generic Letter (GL) 2003-01, "Control Room Habitability" for the Vermont Yankee Nuclear Power Station (ADAMS Accession No. ML051750348). In that letter, Entergy reported the results of the American Society for Testing Materials (ASTM E741, Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution) tracer gas tests for the Vermont Yankee Nuclear Power Station control room that were conducted in 1982. It was determined that the most limiting unfiltered in-leakage into the Control Room Envelope (CRE), was 21.5 cfm, which was less than the value of 3700 cfm assumed in the fuel handling accident radiological analyses for this submittal.

With the assessment made in Section 3.3 of this safety evaluation and the conservative unfiltered in-leakage into the Control Room Envelope, as assumed in the FHA radiological analyses, the NRC staff finds that there is reasonable assurance that the VY control room will be habitable during an FHA with the proposed changes to the containment TSs.

3.5 Summary

The NRC staff reviewed the assumptions, inputs, and methods used by Entergy to assess the radiological impacts of the proposed changes. In conducting this review, the staff relied upon information placed on the docket by Entergy, staff experience in conducting similar reviews, and the staff's confirmatory calculations. The staff finds that Entergy's proposed changes are in accordance with the analysis methods and assumptions consistent with the guidance of RG 1.183 and the proposed TS changes. The staff compared the doses estimated by Entergy to the applicable criteria and to the results of the staff's confirmatory analyses. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses due to postulated DBAs at VY will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. The staff also finds that there is reasonable assurance that the plant design, as modified by this proposed change, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events, and to compensate for uncertainties in accident progression, and in analysis assumptions and parameters. Therefore, the NRC staff concludes that the proposed changes are 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. The NRC staff has determined that the amendments involve no significant increase in the 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, as published in the *Federal Register* on September 16, 2014 (79 FR 55511). 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.

7.0 REFERENCES

1. Wamser, Christopher J., Site Vice President, Entergy Nuclear Operations, Inc., letter to Document Control Desk, U.S. Nuclear Regulatory Commission (NRC), "Technical Specifications Proposed Change No. 306 – Eliminate Certain ESF Requirements during Movement of Irradiated Fuel – Vermont Yankee Nuclear Power Station, BVY-13-097, November 14, 2013, Agencywide Documents Access and Management System (ADAMS) Accession No. ML13323A516.

2. Wamser, Christopher J., Site Vice President, Entergy Nuclear Operations, Inc., letter to Document Control Desk, U.S. Nuclear Regulatory Commission (NRC), "Technical Specifications Proposed Change No. 306 – Eliminate Certain ESF Requirements during Movement of Irradiated Fuel – Supplement 1 - Vermont Yankee Nuclear Power Station, BVY-14-036, June 09, 2014, ADAMS Accession No. ML14163A008.

3. Wamser, Christopher J., Site Vice President, Entergy Nuclear Operations, Inc., letter to Document Control Desk, U.S. Nuclear Regulatory Commission (NRC), "Technical Specifications Proposed Change No. 306 – Eliminate Certain ESF Requirements during Movement of Irradiated Fuel – Supplement 2 - Vermont Yankee Nuclear Power Station, BVY-14-059, August 6, 2014, ADAMS Accession No. ML14224A012.

4. Wamser, Christopher J., Site Vice President, Entergy Nuclear Operations, Inc., letter to Document Control Desk, U.S. Nuclear Regulatory Commission (NRC), "Technical Specifications Proposed Change No. 306 – Eliminate Certain ESF Requirements during Movement of Irradiated Fuel – Supplement 3 - Vermont Yankee Nuclear Power Station, BVY-14-071, October 9, 2014, ADAMS Accession No. ML14288A317.

5. *Technical Specification Task Force*, "Response to NRC Comments and Submittal of Revision 2 of TSTF-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," dated October 11, 2010 and October 19, 2011, ADAMS Accession Nos. ML102850220 and ML110610350.

6. U.S. Nuclear Regulatory Commission, Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined Licenses and Proposed No Significant Hazards Consideration Determination, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont No Significant Hazards Consideration," *Federal Register,* Vol. 79, No. 179, September 16, 2014, pp. 55511 (79 FR 55511).

7. *The Atomic Energy Act of 1954, as Amended, Section 161A* (42 United States Code (U.S.C.) § 2201a) August 8, 2005.

8. *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities," Section 50.36, Technical specifications," Title 10, "Energy" (10 CFR 50.36).

9. *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities,Section 50.36, Technical specifications," Title 10, "Energy," 10 CFR 50.36(c)(2)(ii).

10. *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities," Section 50.67, "Accident source term," Title 10, "Energy," 10 CFR 50.67.

11. *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities, Appendix A to Part 50, "General Design Criteria" Title 10, "Energy," 10 CFR 50, Appendix A.

12. *NRC Regulatory Guide 1.183,* "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000, ADAMS Accession No. ML003716792.

13. NUREG-0800, Standard Review Plan, SRP-15.0.1, Rev. 0, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000, ADAMS Accession No. ML003734190.

14. Letter, USNRC to Entergy Nuclear Operations, Inc. "Vermont Yankee Nuclear Power Station, Issuance of Amendment No. 223, Re: Alternative Source Term (TAC No. MC0253)," NVY-04-045, March 29, 2005, ADAMS Accession No. ML041280490.

15. Letter, USNRC to Technical Specifications Task Force, "Potential Issues with Plant-Specific Adoption of Travelers TSTF-51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations…" November 7, 2013, ADAMS Accession No. ML13246A358.

16. *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities,Section 50.82, "Termination of license," Technical specifications," Title 10, "Energy," 10 CFR 50.82(a)(1)(i) and (ii).

17. General Electric Standard Application for Reactor Fuel, GESTAR II (Supplement for the United States) Licensing Topical Report, pages US-25 through US-28, NEDE-24011-P-A-14-US, Class III14, June 2000.

18. G. Burley, "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," NRC Technical Paper, October 1971, ADAMS Accession No. ML8402080322.

19. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980, ADAMS Accession No. ML070250180.

20. Letter, USNRC to Entergy Nuclear Operations, Inc. "Vermont Yankee Nuclear Power Station, Issuance of Amendment No. 29, Re: Alternative Source Term (TAC No. MC0253)," NVY-04-045, January 28, 1977, ADAMS Accession No. ML011590477.

21. Updated Final Safety Analysis Report (UFSAR) to Entergy Nuclear Operations, Inc. Vermont Yankee Nuclear Power Station, Revision 20, July 1, 2006, ADAMS Accession No. ML061880426.

22. Letter, USNRC to Entergy Nuclear Operations, Inc. "Vermont Yankee Nuclear Power Station, Safety Evaluation, Re: Control of Heavy Loads (Phase I)," NVY-84-139, June 27, 1984.

23. Letter, USNRC to Entergy Nuclear Operations, Inc. "Vermont Yankee Nuclear Power Station, Issuance of Amendment No. 239, Re: Relocation of Reactor Building Crane Technical Specification," July 13, 2009, ADAMS Accession No. ML0191740040.

24. Thayer, Jay K. Site Vice President, Entergy Nuclear Operations, Inc. and Entergy Nuclear Vermont Yankee, LLC, letter to Document Control Desk, U.S.NRC, NRC Generic Letter 2003-01, "Control Room Habitability" – Final Response, BVY-05-067, June 21, 2005, ADAMS Accession No. ML051750348.

25. *U.S. Code of Federal Regulations, Part 20,* -"Standard for Protection Against Radiation," 10 CFR 20.

26. *U.S. Code of Federal Regulations,* Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Function," Section 51.22(c)(9), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Title 10, "Energy," 10 CFR 51.22(c)(9).

27. *U.S. Code of Federal Regulations,* Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Function," Section 51.22(b), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Title 10, "Energy," 10 CFR 51.22(b).

Principal Contributors: M. Blumberg, NRR
S. Peng, NRR
S. Jones, NRR

Date: February 12, 2015

February 12, 2015

Vice President, Operations
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 TO RENEWED FACILITY OPERATING LICENSE RE: ELIMINATE OPERABILITY REQUIREMENTS FOR SECONDARY CONTAINMENT WHEN HANDLING SUFFICIENTLY DECAYED IRRADIATED FUEL OR A FUEL CASK (TAC NO. MF3068)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 262 to Renewed Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 14, 2013, as supplemented by letters dated June 9, 2014, August 6, 2014, and October 9, 2014.

The amendment revises the technical specifications to eliminate the operability requirements for secondary containment when handling sufficiently decayed irradiated fuel or a fuel cask following a minimum of 13 days after the permanent cessation of reactor operation.

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

Sincerely,
/RA/
James Kim, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures:
1. Amendment No. 262 to DPR-28
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
PUBLIC
RidsNrrDorlLpl4-2 Resource
RidsAcrsAcnw_MailCTR Resource
RidsNrrPMVermontYankee Resource
MBlumberg, NRR
RidsNrrLAPBlechman Resource
LPLIV-2 r/f
RidsNrrDorlDpr Resource
R. McKinley, RI
RidsNrrDraArcb Resource
SPeng, NRR
RidsNrrDssStsb Resource
RidsRgn1MailCenter Resource
RidsNrrLAKGoldstein Resource
RidsNrrDssSbpb Resource
RidsNrrDssScvb Resource
SJones, NRR

ADAMS Accession No.: ML14304A588 *by email dated

OFFICE	DORL/LPL4-2/PM	DORL/LPL4-2/LA	DORL/LPL1-1/LA*	DRA/ARCB/BC	DSS/SCVB/BC
NAME	JKim	PBlechman	KGoldstein	UShoop*	RDennig*
DATE	12/1/14	11/4/14	11/25/14	10/03/14	10/15/14
OFFICE	DSS/SBPB/BC	DSS/STSB/BC	OGC/NLO	DORL/LPL4-2/BC	DORL/LPL4-2/PM
NAME	GCasto	RElliott	MSpencer	MKhanna	JKim
DATE	12/1/14	12/2/14	12/5/14	2/11/15	2/12/15

OFFICIAL RECORD COPY