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

October 7, 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 FOR DEFUELED TECHNICAL SPECIFICATIONS AND REVISED LICENSE CONDITIONS FOR PERMANENTLY DEFUELED CONDITION (CAC NO. MF3714)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 263 to Renewed Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station (VY), in response to your application dated March 28, 2014, as supplemented by letters dated April 24, June 9, June 11, and August 13, 2014; and May 4, 2015.

The amendment revises the renewed facility operating license and the associated technical specifications to conform to the permanent shutdown and defueled status at VY.

A copy of the related 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. 263 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. 263
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 March 28, 2014, as supplemented by letters dated April 24, June 9, June 11, and August 13, 2014; and May 4, 2015, 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, Renewed Facility Operating License No. DPR-28 is hereby amended to read, as follows, as indicated in the attachment to this license amendment.

   Paragraph a, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

   a. This paragraph deleted by Amendment No. 263.

   Paragraph b, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

   b. The facility is prohibited from operating the reactor in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

   Paragraph c, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

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

   Paragraph d, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

   d. Entergy Nuclear Vermont Yankee, LLC is financially qualified and Entergy Nuclear Operations, Inc. is technically and financially qualified to engage in the activities authorized by this license, in accordance with the rules and regulations of the Commission; and

   Paragraph f, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

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

   Paragraph g, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

   g. After weighing the environmental, economic, technical and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this license (subject to the conditions for protection of the environment set forth herein) is in accordance with 10 CFR Part 51, of the Commission's regulations and all applicable requirements of said Part 51 have been satisfied; and

Paragraph h, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

h. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

Paragraph 2.A, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

A. Pursuant to Sections 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Entergy Nuclear Vermont Yankee, LLC to possess and use, and Entergy Nuclear Operations, Inc., to possess and use the facility as a utilization facility at the designated location on the Entergy Nuclear Vermont Yankee, LLC site.

Paragraph 2.B, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

B. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as supplemented and amended.

Paragraph 2.C, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

C. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources that was used for reactor startup, sealed sources that was used for calibration of reactor instrumentation and are used in radiation monitoring equipment, and as fission detectors in amounts as required.

Paragraph 3.A, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

A. This paragraph deleted by Amendment No. 263.

Paragraph 3.B, of the 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. 263 are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

Paragraph 3.F, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

F. This paragraph deleted by Amendment No. 263.

Paragraph 3.K, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

K. This paragraph deleted by Amendment No. 263.

Paragraph 3.L, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

L. This paragraph deleted by Amendment No. 263.

Paragraph 3.M, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

M. This paragraph deleted by Amendment No. 263.

Paragraph 3.O, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

O. This paragraph deleted by Amendment No. 263.

Paragraph 3.R, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

R. This paragraph deleted by Amendment No. 263.

Paragraph 3.S, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

S. This paragraph deleted by Amendment No. 263.

Paragraph 4, of the Renewed Facility Operating License No. DPR-28 is hereby amended to read as follows:

4. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena Khanna

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

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: October 7, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 263

RENEWED FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Renewed Facility Operating License No. DPR-28 and 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

Renewed Facility Operating License pages 1 through 14
Cover page for Appendix A
Appendix A, TABLE OF CONTENTS pages i, ii, iii, iv
Appendix A, pages 1 through 267

INSERT

Renewed Facility Operating License pages 1 through 9
Cover page for Appendix A
Appendix A, TABLE OF CONTENTS page i
Appendix A, pages 1 through 20

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.

(Vermont Yankee Nuclear Power Station)

Docket No. 50-271

Renewed Facility Operating License

Renewed Operating License No. DPR-28

The U.S. Nuclear Regulatory Commission (NRC or the Commission), having previously made the findings set forth in Facility Operating License No. DPR-28, dated February 28, 1973, has now found that:

a. This paragraph deleted by Amendment No. 263.

b. The facility is prohibited from operating the reactor in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

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

d. Entergy Nuclear Vermont Yankee, LLC is financially qualified and Entergy Nuclear Operations, Inc. is technically and financially qualified to engage in the activities authorized by this license, in accordance with the rules and regulations of the Commission; and

e. Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations; and

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

g. After weighing the environmental, economic, technical and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this license (subject to the conditions for protection of the environment set forth herein) is in accordance with 10 CFR Part 51, of the Commission's regulations and all applicable requirements of said Part 51 have been satisfied; and

h. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

Accordingly, Facility Operating License No. DPR-28, as amended, issued to Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. is superseded by Renewed Facility Operating License No. DPR-28 and is hereby amended in its entirety to read:

1. This renewed license applies to the Vermont Yankee Nuclear Power Station (the facility), a single cycle, boiling water, light water moderated and cooled reactor, and associated electric generating equipment. The facility is located on Entergy Nuclear Vermont Yankee, LLC's site, in the Town of Vernon, Windham County, Vermont, and is described in the application as amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

   A. Pursuant to Sections 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Entergy Nuclear Vermont Yankee, LLC to possess and use, and Entergy Nuclear Operations, Inc., to possess and use the facility as a utilization facility at the designated location on the Entergy Nuclear Vermont Yankee, LLC site.

   B. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as supplemented and amended.

   C. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment, and as fission detectors in amounts as required.

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. This paragraph deleted by Amendment No. 263.

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 263, 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.

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

4. If harmful effects or evidence of irreversible damage in land or water ecosystems as a result of facility operation are detected by Entergy Nuclear Operations, Inc.'s environmental monitoring program, Entergy Nuclear Operations, Inc. shall provide an analysis of the problem to the Commission and to the advisory group for the Technical Specifications, and Entergy Nuclear Operations, Inc. thereafter will provide, subject to the review by the aforesaid advisory group, a course of action to be taken immediately to alleviate the problem.

5. Entergy Nuclear Operations, Inc. will grant authorized representatives of the Massachusetts Department of Public Health (MDPH) and Metropolitan District Commission (MDC) access to records and charts related to discharge of radioactive materials to the Connecticut River.

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

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

8. Entergy Nuclear Operations, Inc. will permit authorized representatives of the MDPH and MDC to examine the chemical and radioactivity analyses performed by Entergy Nuclear Operations, Inc.

9. Entergy Nuclear Operations, Inc. shall immediately notify MDPH, or an agency designated by MDPH, in the event concentrations of radioactive materials in liquid effluents, measured at the point of release from the Vermont Yankee facility, exceed the limit set forth in the facility Offsite Dose Calculation Manual. Entergy Nuclear Operations, Inc. will also notify MDPH in writing within 30 days following the release of radioactive materials in liquid effluents in excess of 10 percent of the limit set forth in the facility Offsite Dose Calculation Manual.

10. A report shall be submitted to MDPH and MDC by May 15 of each year of plant operation, specifying the total quantities of radioactive materials released to the Connecticut River during the previous calendar year. The report shall contain the following information:

    (a) Total curie activity discharged other than tritium and dissolved gases.

    (b) Total curie alpha activity discharged.

    (c) Total curies of tritium discharged.

    (d) Total curies of dissolved radio-gases discharged.

(e) Total volume (in gallons) of liquid waste discharged.

(f) Total volume (in gallons) of dilution water.

(g) Average concentration at discharge outfall.

(h) This paragraph deleted by Amendment No. 206, October 22, 2001.

(i) Total radioactivity (in curies) released by nuclide including dissolved radio-gases.

(j) Percent of the facility Offsite Dose Calculation Manual limit for total activity released.

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

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

13. Entergy Nuclear Operations, Inc. shall establish and maintain a system of emergency notification to the states of Vermont and New Hampshire, and the Commonwealth of Massachusetts, satisfactory to the appropriate public health and public safety officials of those states and the Commonwealth, which provides for:

    a. Notice of site emergencies as well as general emergencies.

    b. Direct microwave communication with the state police headquarters of the respective states and the Commonwealth when the transmission facilities of the respective states and the Commonwealth so permit, at the expense of Entergy Nuclear Operations, Inc.

    c. A verification or coding system for emergency messages between Entergy Nuclear Operations, Inc. and the state police headquarters of the respective states and the Commonwealth.

14. Entergy Nuclear Operations, Inc. shall furnish advance notification to MDPH, or to another Commonwealth agency designated by MDPH, of the time, method and proposed route through the Commonwealth of any shipments of nuclear fuel and wastes to and from the Vermont Yankee facility which will utilize railways or roadways in the Commonwealth.

F. This paragraph deleted by Amendment No. 263.

G. Security Plan

Entergy Nuclear Operations, Inc. shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Vermont Yankee Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 0," submitted by letter dated October 18, 2004, as supplemented by letter dated May 16, 2006.

Entergy Nuclear Operations, Inc. shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). Entergy Nuclear Operations, Inc. CSP was approved by License Amendment No. 247, as supplemented by changes approved by License Amendment Nos. 251 and 259.

H. This paragraph deleted by Amendment No. 107, 8/25/88.

I. This paragraph deleted by Amendment No. 131, 10/7/91.

J. License Transfer Conditions

On the closing date of the transfer of Vermont Yankee Nuclear Power Station (Vermont Yankee), Entergy Nuclear Vermont Yankee, LLC shall obtain from Vermont Yankee Nuclear Power Corporation all of the accumulated decommissioning trust funds for the facility, and ensure the deposit of such funds into a decommissioning trust for Vermont Yankee established by Entergy Nuclear Vermont Yankee, LLC. If the amount of such funds does not meet or exceed the minimum amount required for the facility pursuant to 10 CFR 50.75, Entergy Nuclear Vermont Yankee, LLC shall at such time deposit additional funds into the trust and/or obtain a parent company guarantee (to be updated annually) and/or obtain a surety pursuant to 10 CFR 50.75(e)(1)(iii) in a form acceptable to the NRC and in an amount or amounts which, when combined with the decommissioning trust funds for the facility that have been obtained and deposited as required above, equals or

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

exceeds the total amount required for the facility pursuant to 10 CFR 50.75. The decommissioning trust, and surety if utilized, shall be subject to or be consistent with the following requirements, as applicable:

a. Decommissioning Trust

(i) The decommissioning trust agreement must be in a form acceptable to the NRC.

(ii) With respect to the decommissioning trust funds, investments in the securities or other obligations of Entergy Corporation and its affiliates, successors, or assigns shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(iii) The decommissioning trust agreement must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the trustee has first given the NRC 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the Director of the Office of Nuclear Reactor Regulation.

(iv) The decommissioning trust agreement must provide that the agreement cannot be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

(v) The appropriate section of the decommissioning trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations.

b. Surety

(i) The surety agreement must be in a form acceptable to the NRC and be in accordance with all applicable NRC regulations.

(ii) The surety company providing any surety obtained to comply with the Order approving the transfer shall be one of those listed by the U.S. Department of the Treasury in the most recent edition of Circular 570 and shall have a coverage limit sufficient to cover the amount of the surety.

(iii) Entergy Nuclear Vermont Yankee, LLC shall establish a standby trust to receive funds from the surety, if a surety is obtained, in the event that Entergy Nuclear Vermont Yankee, LLC defaults on its funding obligations for the decommissioning of Vermont Yankee. The standby trust agreement must be in a form acceptable to the NRC, and shall conform with all conditions otherwise applicable to the decommissioning trust agreement.

(iv) The surety agreement must provide that the agreement cannot be amended in any material respect, or terminated, without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

Entergy Nuclear Vermont Yankee, LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of this license to Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. shall take no action to cause Entergy Global Investments, Inc., or Entergy International Holdings Ltd. LLC, or their parent companies to void, cancel, or modify the lines of credit to provide funding for Vermont Yankee as represented in the application without prior written consent of the Director of the Office of Nuclear Reactor Regulation.

K. This paragraph deleted by Amendment No. 263.

L. This paragraph deleted by Amendment No. 263.

M. This paragraph deleted by Amendment No. 263.

N. <u>Mitigation Strategy License Condition</u>

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications

3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

O. This paragraph deleted by Amendment No. 263.

P. The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application process, and as supplemented by Commitment Nos. 1-5, 6 (as revised by Entergy Nuclear Vermont Yankee, LLC letter dated May 19, 2011), 7-36, 38, 39, 42, 43, and 45-55 of Appendix A of Supplement 2 of NUREG-1907 shall be incorporated as part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. may make changes to the programs and activities described in the UFSAR supplement and Commitment Nos. 1-5, 6 (as revised by Entergy Nuclear Vermont Yankee, LLC letter dated May 19, 2011), 7-36, 38, 39, 42, 43, and 45-55 of Appendix A of Supplement 2 of NUREG-1907 provided Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

Q. This paragraph deleted by Amendment No. 256, April 17, 2013.

R. This paragraph deleted by Amendment No. 263.

S. This paragraph deleted by Amendment No. 263.

4. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
Eric J. Leeds

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosures:
Appendix A - Technical Specifications

Date of Issuance: March 21, 2011

APPENDIX A

TO

OPERATING LICENSE DPR-28

TECHNICAL SPECIFICATIONS

AND BASES

FOR

VERMONT YANKEE NUCLEAR POWER STATION

VERNON, VERMONT

ENTERGY NUCLEAR OPERATIONS, INC.

AND

ENTERGY NUCLEAR VERMONT YANKEE, LLC

DOCKET NO. 50-271

TABLE OF CONTENTS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Certified Fuel Handler - A Certified Fuel Handler is an individual who complies with the provisions of the Certified Fuel Handler training program.

B. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

C. Operable - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal or emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

D. Operating - Operating means that a system or component is performing its intended functions in its required manner.

3.0 LIMITING CONDITIONS FOR OPERATION APPLICABILITY

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 4.0.1

SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified frequency shall be failure to meet the LCO except as provided in SR 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 4.0.2

Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%.

SR 4.0.3

If it is discovered that a surveillance was not performed within its specified frequency, declaring applicable Limiting Conditions for Operation (LCOs) not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

3.0 LIMITING CONDITIONS FOR OPERATION APPLICABILITY

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 4.0.3 (Continued)

If the surveillance is not performed within the delay period, applicable LCOs must immediately be declared not met, and applicable LCOs must be entered.

When the surveillance is performed within the delay period and the surveillance is not met (i.e., acceptance criteria are not satisfied), applicable LCOs must immediately be declared not met, and applicable LCOs must be entered.

TS 4.0 Surveillance Requirement (SR) Applicability

SR 4.0.1 Bases

SR 4.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that variables are within specified limits. Failure to meet a Surveillance within the specified frequency, in accordance with SR 4.0.2, constitutes a failure to meet an LCO.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR.

SR 4.0.2 Bases

SR 4.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers unit conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the surveillance at its specified frequency. This is based on the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 4.0.2 are those Surveillances for which the 25% extension of the interval specified in the frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS.

The provisions of SR 4.0.2 are not intended to be used repeatedly merely as an operational convenience to extend surveillance intervals (other than those consistent with refueling intervals).

SR 4.0.3 Bases

SR 4.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the surveillance has not been performed in accordance with SR 4.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete surveillances that have been missed. This delay period permits the completion of a surveillance before complying with action statements or other remedial measures that might preclude completion of the Surveillance.

SR 4.0.3 Bases (Continued)

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

Failure to comply with specified surveillance frequencies is expected to be an infrequent occurrence. Use of the delay period established by SR 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel and the time required to perform the surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the licensee's Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the variable is considered outside the specified limits and the completion times of the Action Statements for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the variable is outside the specified limits and the completion times of the Action Statements for the applicable LCO Conditions begin immediately upon the failure of the surveillance.

Completion of the surveillance within the delay period allowed by this Specification, or within the completion time of the ACTIONS, restores compliance with SR 4.0.1.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the release of all radioactive effluents from the plant.

Objective:

To assure that radioactive effluents are kept "as low as is reasonably achievable" in accordance with 10CFR50, Appendix I and, in any event, are within the dose limits for Members of the Public specified in 10CFR20.

Specification:

A. Liquid Holdup Tanks

1. The quantity of radioactive material contained in any outside tank* shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

2. With the quantity of radioactive material in any outside tank* exceeding the limit of Specification 3.1.A.1, immediately take action to suspend all additions of radioactive material to the tank. Within 48 hours, reduce the tank contents to within the limit.

4.1 SURVEILLANCE REQUIREMENTS

4.1 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the required surveillance of all radioactive effluents released from the plant.

Objective:

To ascertain that all radioactive effluents released from the plant are kept "as low as is reasonably achievable" in accordance with 10CFR50, Appendix I and, in any event, are within the dose limits for Members of the Public specified in 10CFR20.

Specification:

A. Liquid Holdup Tanks

1. The quantity of radioactive material contained in each of the liquid holdup tanks* shall be determined to be within the limits of Specification 3.1.A.1 by analyzing a representative sample of the tank's contents within one week following the addition of radioactive materials to the tank. One sample may cover multiple additions.

*NOTE: Tanks included in this Specification are only those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank's contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

BASES:

3.1 RADIOACTIVE EFFLUENTS

A. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an Unrestricted Area.

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 SPENT FUEL STORAGE

Applicability:

Applies to storage of spent fuel.

Objective:

To assure safe storage of spent fuel.

Specification:

A. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool the pool water level shall be maintained at a level of at least 36 feet.

B. 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.

4.2 SURVEILLANCE REQUIREMENTS

4.2 SPENT FUEL STORAGE

Applicability:

Applies to the parameters which monitor the storage of spent fuel.

Objective:

To verify that spent fuel is being stored safely.

Specification:

A. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.

B. 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 operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.

BASES:

3.2 & 4.2 SPENT FUEL STORAGE

A. To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. This minimum water level of 36 feet is established because it would be a significant change from the normal level, well above a level to assure adequate cooling (just above active fuel).

B. The Spent Fuel Pool Cooling System is designed to maintain the pool water temperature below 125°F during normal operations. If the reactor core is completely discharged, the temperature of the pool water may increase to greater than 125°F.

5.0 DESIGN FEATURES

5.1 Site

The station is located on the property on the west bank of the Connecticut River in the Town of Vernon, Vermont, which Entergy Nuclear Vermont Yankee, LLC either owns or to which it has perpetual rights and easements. The site plan showing the exclusion area boundary, boundary for gaseous effluents, boundary for liquid effluents, as well as areas defined per 10CFR20 as "controlled areas" and "unrestricted areas" are on plant drawing 5920-6245. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 is 910 feet.

The licensee will at all times retain the complete authority to determine and maintain sufficient control of all activities through ownership, easement, contract and/or other legal instruments on property which is closer to the reactor center line than 910 feet. This includes the authority to exclude or remove personnel and property within the exclusion area. Only activities related to plant operation are permitted in the exclusion area.

5.2 Spent Fuel Storage

A. The K_{eff} of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

B. Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

C. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3353.

D. The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool or the new fuel storage facility shall be less than or equal to 1.31 at 20°C.

BASES:

5.1 SITE

Exclusion area means that area surrounding the reactor, as measured from the reactor center line, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided those are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of an emergency, to protect the public health and safety.

Contract provisions for property agreements in the exclusion area will ensure that the licensee retains sufficient control of all activities in the exclusion area including the authority to exclude or remove personnel and property, thereby (1) maintaining compliance with 10CFR50.67 radiological limits for the exclusion area, and (2) ensuring that any and all activities, now or in the future, in the exclusion area would not negatively affect nuclear safety, safe plant operation or violate current plant design or licensing basis.

Any property transaction in the exclusion area, as is the case for any activity which has the potential to adversely affect nuclear safety or safe plant operation, requires a review in accordance with 10CFR50.59. Additionally, any property transaction would be required to comply with other regulatory requirements (e.g., 10CFR50.83) as applicable.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The plant manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during absences.

B. The plant manager or designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

C. The shift supervisor shall be responsible for the shift command function.

6.2 ORGANIZATION

A. Onsite and Offsite Organizations

Organizations shall be established for facility staff and corporate management. These organizations shall include the positions for activities affecting safety of the nuclear fuel.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Program Manual. The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Technical Requirements Manual.

2. The plant manager shall be responsible for overall facility safe operation and shall have control over those on-site activities necessary for safe storage and maintenance of the nuclear fuel.

3. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure safe management of nuclear fuel.

4. The individuals who train the Certified Fuel Handlers, carry out health physics, or perform quality assurance functions may report to the appropriate on-site manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

6.2 ORGANIZATION (Cont'd)

B. Facility Staff

The facility staff organization shall include the following:

1. Each duty shift shall be composed of at least one shift supervisor and one Non-certified Operator. The Non-certified Operator position may be filled by a Certified Fuel Handler.

   2. At least one person qualified to stand watch in the control room (Non-certified Operator or Certified Fuel Handler) shall be present in the control room when nuclear fuel is stored in the spent fuel pool.

   3. All fuel handling operations shall be directly supervised by a Certified Fuel Handler.

4. Shift crew composition shall meet the requirements stipulated herein. Shift crew composition may be less than the minimum requirement of Specification 6.2.B.1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:

   a. no fuel movements are in progress; and

   b. no movement of loads over fuel are in progress; and

   c. no unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

5. An individual qualified in radiation protection procedures shall be present on-site during the movement of fuel and during the movement of loads over fuel.

6. Deleted

7. The shift supervisor shall be a Certified Fuel Handler.

8. Deleted

C. Facility Staff Qualifications

1. Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the Quality Assurance Program Manual (QAPM).

2. An NRC approved training and retraining program for Certified Fuel Handlers shall be maintained.

6.3 Deleted

6.4 PROCEDURES

Written procedures shall be established, implemented, and maintained covering the following activities:

A. Normal startup, operation and shutdown of systems and components needed for the safe storage of nuclear fuel.

B. Fuel handling operations.

6.4 PROCEDURES (Cont'd)

C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components needed for the safe storage of nuclear fuel.

D. Emergency conditions involving potential or actual release of radioactivity.

E. Preventive and corrective maintenance operations which could have an effect on the safety of the nuclear fuel.

F. Surveillance and testing requirements.

G. Fire protection program implementation.

H. Process Control Program in-plant implementation.

I. Off-Site Dose Calculation Manual implementation.

6.5 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraphs 20.1601(a) and 20.1601(b) of 10 CFR 20:

A. High Radiation Areas with dose rates greater than 0.1 rem/hour at 30 centimeters, but not exceeding 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation:

1. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

2. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

3. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

4. Each individual or group entering such an area shall possess:

a. A radiation monitoring device that continuously displays radiation dose rates in the area, or

b. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

c. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

d. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

1. Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation does rates in the area; who is responsible for controlling personnel exposure within the area, or

2. Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

5. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

B. High Radiation Areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 rads/hour at 1 meter from the radiation source or from any surface penetrated by the radiation:

1. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

a. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, and/or radiation protection manager, or his or her designee.

b. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

2. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation does rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

3. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

4. Each individual or group entering such an area shall possess one of the following:

   a. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   b. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

   c. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

      1. Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

      2. Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

   d. In those cases where option (b) and (c), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

6.6 REPORTING REQUIREMENTS

The following reports shall be submitted in accordance with 10 CFR 50.4.

A. Deleted
B. Deleted
C. Deleted

D. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility shall be submitted by May 15 of each year and in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM) and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

E. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to the table and figures in the ODCM. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

6.7 PROGRAMS AND MANUALS

The following programs shall be established, implemented and maintained:

A. Deleted

B. OFF-SITE DOSE CALCULATION MANUAL (ODCM)

An Off-Site Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents for the purpose of demonstrating compliance with 10 CFR 50, Appendix I, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Radioactive Effluent Release Report and the Annual Radiological Environmental Operating Report required by Specification 6.6.D and Specification 6.6.E, respectively.

6.7 PROGRAMS AND MANUALS (Cont'd)

1. Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

i. Sufficient information to support the change together with appropriate analyses or evaluations justifying the change(s) and

ii. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, and do not adversely impact the accuracy or reliability of effluent dose or setpoint calculations.

b. Shall become effective upon approval by the plant manager.

c. Shall be submitted to the Commission in the form of a legible copy of the affected pages of the ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

C. Deleted

D. Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001 - 20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the facility to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the facility to areas at or beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from the facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

E. TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license, or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59

6.7 PROGRAMS AND MANUALS (Cont'd)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.7.E.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 263

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-28

ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated September 23, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13273A204), Entergy Nuclear Operations, Inc. (ENO, the licensee), submitted a notification to the U.S. Nuclear Regulatory Commission (NRC) indicating that it would permanently shut down Vermont Yankee Nuclear Power Station (VY) in the fourth calendar quarter of 2014. On December 29, 2014, ENO permanently ceased power operations at VY. On January 12, 2015, ENO certified that it had permanently defueled the VY reactor vessel (ADAMS Accession No. ML15013A426). Consequently, pursuant to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.82(a)(2), the VY renewed facility operating license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

By letter dated March 28, 2014 (ADAMS Accession No. ML14091A291), as supplemented by letters dated April 24, June 9, June 11, and August 13, 2014; and May 4, 2015 (ADAMS Accession Nos. ML14119A101, ML14163A009, ML14168A282, ML14231A017, and ML15127A171, respectively), ENO requested an amendment to revise the VY renewed facility operating license and the associated technical specifications (TSs) to permanently defueled TSs consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

The supplemental letters dated April 24, June 9, June 11, August 13, 2014; and May 4, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 17, 2015 (80 FR 8358).

Enclosure 2

2.0 REGULATORY EVALUATION

2.1 Technical Specifications

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the application. The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." Pursuant to 10 CFR 50.36, each operating license issued by the Commission includes TSs and includes items in the following categories: (1) safety limits (SLs), limiting safety systems settings and control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

Section 50.36 of 10 CFR provides four criteria to define the scope of equipment and parameters to be included in the TS LCOs. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the reactor coolant system (RCS) pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during design-bases accidents (DBAs) or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those structures, systems, and components (SSCs) shown to be significant to public health and safety.

A general discussion of the criteria that were used by the NRC staff in its evaluation to ensure that TS LCOs proposed for deletion are no longer required to be included in TSs, is provided below.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for "installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary." Since no fuel is present in the reactor or RCS at the VY facility, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established 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." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. The scope of DBAs applicable to a reactor permanently shutdown and defueled is reduced from those postulated for an operating reactor, and most TSs satisfying Criterion 2 are no longer applicable. One existing TS that defines the initial condition of the DBA associated with irradiated fuel movement is discussed in Section 3.5 of this safety evaluation (SE).

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs for operation must be established for a SSC "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." The intent of this criterion is to capture into TSs those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. There are no transients that continue to apply to permanently shutdown and defueled reactors. The scope of applicable DBAs that apply to VY is discussed in more detail in Section 3.0 of this SE.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs "which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. There are no longer any DBAs apply to permanently shutdown and defueled condition at VY that can result in a significant offsite radiological risk to public health and safety.

The NRC staff notes that in the course of this evaluation, information contained in NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," Draft Report for Comment, March 1998 (ADAMS Accession No. ML082330233), was also considered. This draft NUREG provides examples of TSs that the staff found acceptable during previous TS reviews for permanently shutdown and defueled reactors.

2.2 Radiological Consequences

Section 14 of the VY Updated Final Safety Analysis Report (UFSAR), "Station Safety Analysis" describes the DBA scenarios that are applicable to VY during power and refueling operations and the accidents with the greatest potential for radiation exposure. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products. When the reactor is permanently defueled and irradiated fuel assemblies are stored in the spent fuel pool (SFP) and the independent spent fuel storage installation (ISFSI), the spectrum of credible accidents is much smaller than for an operational plant, and most of the accident scenarios postulated in the UFSAR are no longer possible.

The licensee stated that the only accident with potential offsite radiological consequences that remains applicable to VY in the permanently shutdown and defueled condition is a fuel handling accident (FHA) in the reactor building where the SFP is located. The FHA analysis for VY shows that following 13 days of decay time after reactor shutdown, the dose consequences from an FHA are acceptable without certain systems operable during and following the event, provided that 20.67 feet of water is maintained above the irradiated fuel assemblies in the SFP.

The NRC staff evaluated the radiological consequences of the postulated FHA DBA against the dose criteria specified in 10 CFR 50.67, "Accident source term," and using the guidance described in NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792). RG 1.183 provides guidance to licensees on acceptable application of alternative source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

By letter dated March 29, 2005, the NRC approved the implementation of the AST methodology for FHA dose consequence analysis at VY by License Amendment No. 223 to Renewed Facility Operating License DPR-28 (ADAMS Accession No. ML041280490). The submittal also included changes to the VY TSs to reflect implementation of AST assumptions in accordance with 10 CFR 50.67.

The FHA-specific dose acceptance criteria are specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," (SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190). The dose acceptance criteria for the FHA are a total effective dose equivalent (TEDE) of 6.3 roentgen equivalent man (rem) at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room (CR) for the duration of the accident.

3.0 TECHNICAL EVALUATION

3.1 Accident Analysis

During normal power reactor operations, the forced inlet flow of water through the RCS removes the heat from the reactor by generating steam. The steam system, operating at high temperatures and pressures, transfers this heat to the turbine generator. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the RCS. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems, which could affect the reactor core. With the termination of reactor operations at VY and the permanent removal of the fuel from the reactor core, most of the DBAs postulated in the UFSAR will no longer be possible. The irradiated fuel will be stored in the SFP and the ISFSI. The reactor, RCS, steam system, and turbine generator are no longer in operation and have no function related to the storage of the irradiated fuel. Therefore, the postulated accidents involving failure or malfunction of the reactor, RCS, steam system, or turbine generator are no longer applicable.

Prior to permanently shutting down, Section 14 of the VY UFSAR described transients and DBAs that were applicable to VY during power operations. Since VY is permanently shut down and defueled, there are no longer any applicable transients. The only DBA that could potentially apply at this time to the permanently shutdown and defueled VY reactor would be the FHA.

3.2 Fuel Handling Accident

Post-cessation of operations, an FHA onto the top of the core (or elsewhere within containment) is no longer possible and therefore, no longer part of the licensing basis. However, an FHA in the SFP (which is located in the reactor building) is still possible at VY, as long as spent fuel is stored in the SFP.

A revision to the FHA analysis was performed by the licensee to support the license amendment request "Eliminate Operability Requirements for Secondary Containment when Handling Sufficiently Decayed Irradiated Fuel or a Fuel Cask," dated November 14, 2013 (ADAMS Accession No. ML13323A516) to address the permanently defueled condition. The analysis assumed a reasonable time post-cessation of operations for movement of fuel. The licensee then determined that if an FHA occurs, the dose consequences would be within 10 CFR 50.67 and RG 1.183 dose limits. The licensee evaluated the days of decay time after the reactor had been shut down in which the dose consequences were acceptable without relying on safety-related SSCs remaining operable for accident mitigation during and following the event. The licensee evaluated the maximum 2-hour TEDE to an individual located at the EAB, the 30-day TEDE to an individual at the outer boundary of the LPZ, and the CR. The resulting doses were less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria, and the 10 CFR 50.67 limits.

The revised FHA analysis involves the inadvertent dropping of a fuel assembly during fuel handling operations, and the subsequent rupture of fuel rods in the dropped assembly, and/or the impacted assembly. The accident is assumed to involve the dropping of an assembly onto other assemblies, leading to the equivalent clad failure of the fuel rods. The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously into the fuel storage pool. The FHA dose analysis models 13 days of radioactive decay prior to the event. The NRC staff finds that the decay time assumed by the licensee is consistent with RG 1.183, Regulatory Position 3.1 and to be conservative because VY has been defueled since January 12, 2015, which is much greater than 13 days as of today. Based on the actual time to date for spent fuel decay, the iodine source terms would be negligible.

Fission products released from the damaged fuel are decontaminated by passage through the pool water, with the degree of decontamination dependent upon their physical and chemical form. The licensee assumed no decontamination for noble gases, a decontamination factor of 200 for radioiodine, and retention of all aerosol and particulate fission products. In ENO's response dated October 9, 2014, to the NRC staff's request for additional information (ADAMS Accession No. ML14288A317), ENO compared the decontamination factors and number of damaged assemblies for an FHA in the SFP to those for an FHA over the reactor core. ENO stated that the minimum amount of water needed 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. ENO also stated that although the decontamination factor in the SPF is reduced (due to there being 20.67 feet of water above the postulated damaged fuel assembly), this reduction is more than offset by the smaller number of postulated damaged fuel rods in the SFP.

The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously from the fuel and the overlying fuel pool to the secondary containment building. ENO's FHA with an open containment model no longer credits several safety systems after 13 days of fuel decay and is 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, standby gas treatment (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. Therefore, ENO assumes the release to the environment is an unfiltered ground-level release via the reactor building blowout panels.

The NRC staff FHA analysis confirmed that the release point from the reactor building to the control room uses the most limiting combination of release and receptor locations. Since ENO is assuming a ground level release, the NRC staff finds it acceptable for the licensee to model the FHA using the CR, EAB, and LPZ atmospheric dispersion factors for an FHA with a ground level release. These factors were previously approved in License Amendment No. 223.

The licensee's analysis of radiological consequences resulting from the postulated FHA for the permanently shutdown and defueled condition at VY concluded that the radiological consequences at the EAB, LPZ, and in the CR are within the dose criteria specified in 10 CFR 50.67 and accident-specific dose criteria described in SRP Section 15.0.1. The NRC staff has evaluated the licensee's analysis. In performing this evaluation, the NRC staff relied upon information provided by the licensee, as well as NRC staff experience in performing similar evaluations. The staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183.

3.3 Proposed TS Changes

3.3.1 Section 1.0, "Definitions"

The licensee proposed deleting the following definitions from TS 1.0 because they pertain to an operating reactor. Since VY is permanently shut down and defueled, the definitions no longer apply:

A. Reportable Occurrence - The equivalent of a reportable event which shall be any of the conditions specified in Section 50.73 to 10 CFR Part 50.

B. Alteration of the Reactor Core - The act of moving any component affecting reactivity within the reactor vessel in the region above the core support plate, below the upper grid and within the shroud. Normal movement of control rods or neutron detectors, or the replacement of neutron detectors is not defined as a core alteration.

C. Hot Standby - Hot standby means operation with the reactor critical and the main steam line isolation valves closed.

E. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time as specified is not part of the routine instrument calibration but will be checked once per operating cycle.

F. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

G. Instrument Functional Test - An instrument functional test shall be:

1. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

2. Bistable channels - the injection of a signal into the sensor to verify the operability including alarm and/or trip functions.

H. Logic System Functional Test - A logic system functional test shall be a test of all logic components required for operability of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify operability. The logic system functional test may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

I. Minimum Critical Power Ratio - The minimum critical power ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the appropriate NRC-approved critical power correlation to the actual assembly operating power.

J. Mode - The reactor mode is that which is established by the mode-selector-switch.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment, which are no longer required to be open during accident conditions, are closed. Such valves may be opened intermittently under administrative controls.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

O. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1912 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1912 thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser vacuum trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM [average power range monitor] neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig [pounds per square inch gauge] and the reactor protection system is energized with APRM protection and RBM [rod-block monitor] interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

V. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212 °F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

3. Shutdown means conditions as above such that the effective multiplication factor (K_{eff}) of the core shall be less than 0.99.

X. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

BB. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CC. Dose Equivalent I-131 - The dose equivalent I-131 shall be that concentration of I-13I (microcurie/gram) which alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, 1-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988; FGR 12, "External Exposure to Radionuclides In Air, Water, and Soil," 1993; or NRC Regulatory Guide 1.109, Revision 1, October 1977.

NN. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.C. Plant operation within these operating limits is addressed in individual specifications.

OO. Reactor Protection System (RPS) Response Time - RPS Response Time shall be the time from the opening of the sensor contact up to and including the opening of the scram solenoid relay.

The NRC staff examined the TS definitions proposed for deletion and concludes that all the terms listed above are only meaningful to a reactor authorized to operate. Since VY is permanently shut down and defueled, the staff finds the deletion of these definitions from the TSs acceptable.

The licensee proposes to delete the following definitions from TS 1.0 because they are redundant to the information provided in TSs 4.0.1, 4.0.2, and 4.0.3.

Y. Surveillance Frequency - Relocated to Specifications 4.0.2 and 4.0.3.

Z. Surveillance Interval - Relocated to Specification 4.0.1.

The NRC staff examined the TS definitions "Surveillance Frequency" and "Surveillance Interval" proposed for deletion and concludes that these definitions are included in TSs 4.0.1, 4.0.2, and 4.0.3 and therefore, their deletion from the TSs are acceptable.

In addition, the licensee proposed adding a definition for "Certified Fuel Handler." The licensee proposes to define a certified fuel handler as:

A. Certified Fuel Handler – A Certified Fuel Handler is an individual who complies with provisions of the Certified Fuel Handler training program.

TS 6.2.C.2 states "[a]n NRC approved training and retraining program for Certified Fuel Handlers shall be maintained." The NRC staff finds the definition of a Certified Fuel Handler conforms to the usage contained in the Administrative Controls section of the VY permanently defueled TSs and is consistent with the definition in 10 CFR Part 50, and is therefore, acceptable.

3.3.2 TS Sections 1/2.1 and 1/2.2 "Safety Limit" and "Limiting Safety System Setting"

This section establishes SLs, which preclude violation of the fuel cladding integrity and RCS design pressure.

TS Sections 1.1 and 1.2, "Safety Limit" contains two separate specifications:

- TS 1.1, "Fuel Cladding Integrity"
- TS 1.2, "Reactor Coolant System"

SLs for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of the certain physical barriers that guard against the uncontrolled release of radioactivity. The TS SL section also contains TSs 2.1 and 2.2, "Limiting Safety System Setting." Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. The restrictions promulgated in TSs 1.1 and 2.1 prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. TSs 1.1 and 2.1 are applicable in Modes Run and Startup/Hot Standby. TSs 1.2 and 2.2 promulgate requirements on parameters to protect the integrity of the RCS against overpressure. TS 1.2 and 2.2 are applicable in all Modes.

The licensee proposes to delete the SLs and limiting safety system settings specified in TS Sections 1/2.1 and 1/2.2 because they are not applicable to a reactor that is permanently shut down and defueled. The licensee states that the SLs and limiting safety system settings TSs limit important process variables that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the reactor in all Modes. However, 10 CFR 50.82(a)(2) prohibits operation of the reactor or placing fuel in the reactor vessel. Therefore, the SL and limiting safety system settings TSs only address specific process variables that are no longer applicable to VY.

The NRC staff examined the SLs, limiting safety system settings and their Bases. There are two SLs in Section 1/2.1 and 1/2.2: "Fuel Cladding Integrity" and "Reactor Coolant System." The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. The RCS pressure limit is set such that the integrity of the RCS is not threatened due to an overpressure condition. As stated in the Bases, "the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly, for which more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties." A limit is placed on the minimum critical power ratio. The Bases for the maximum RCS pressure state that "reactor coolant system [integrity] is an important barrier in the prevention of an uncontrolled release of fission products." Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations is no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the Commission that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this permanently defueled condition, there will be no minimum critical power ratio to be monitored and there will be no challenge to the RCS integrity. Based on these findings, the staff concludes that the SLs and limiting safety system settings no longer apply.

Based on the above, the NRC staff finds that the proposed deletions of TS Sections 1/2.1 and 1/2.2 are acceptable.

3.3.3 TS Section 3/4.0, "Limiting Conditions for Operations Applicability" and "Surveillance Requirement (SR) Applicability"

This section contains the general requirements applicable to all LCOs and SRs and applies at all times unless otherwise stated in TSs.

LCO 3.0.8, "Inoperability of Snubbers," establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). The licensee proposes to delete LCO 3.0.8 in its entirety because all systems associated with snubbers are no longer required to be operable at VY given it is permanently shut down and in a defueled condition.

SR 4.0.1 establishes the requirement that SRs must be met during the modes or other specified conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification ensures that surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. The licensee proposes to revise this TS to delete the reference to "modes" since the reference to modes is no longer applicable to the VY reactor, which is permanently shut down and defueled.

SR 4.0.2 establishes the requirement that periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals and allows an interval adjustment of 25 percent. In addition, it states that the operating cycle interval is considered to be 18 months. The licensee proposes to delete the sentence that discusses the operating cycle interval because the reference to operating cycle is no longer applicable when the reactor is permanently shut down and defueled at VY.

The NRC staff reviewed the proposed changes to the VY TSs concerning LCO and SR applicability. The staff concludes that the changes are consistent with the transition to a permanently shutdown and defueled facility. The provisions of 10 CFR 50.82(a)(2) prohibit a licensee of a permanently defueled facility from operating the plant or placing fuel in the reactor vessel. The proposed change to delete the references to "mode" is appropriate because the term refers to conditions of normal operation. Since the licensee will no longer be operating, reference to operating modes is no longer needed and the discussion of the operating cycle interval. Similar to modes, operating cycle interval no longer applies because the licensee will no longer be operating. Therefore, the changes appropriately reflect the change in plant status and are acceptable.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, all systems associated with snubbers are no longer required to be operable and are proposed for deletion from the TSs. As such, LCO 3.0.8 no longer applies to any systems remaining in the TSs. Therefore, the proposed deletion of LCO 3.0.8 is acceptable.

Based on the above, the staff finds that the proposed changes to LCO 3.0.8, SR 4.0.1, and SR 4.0.2 are acceptable.

3.3.4 TS Section 3/4.1, "Reactor Protection System (RPS)"

Section 3/4.1 of the VY TSs, "Reactor Protection System (RPS)," contains LCOs, actions, and SRs required for safe operation of the facility. The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary (RCPB) and minimizes the energy that must be absorbed following a loss-of-coolant accident (LOCA). The RPS TS applies to be operable in Run, Startup/Hot Standby and Refuel with the reactor coolant temperature greater than 212 degrees Fahrenheit (F), and in Refuel with reactor coolant temperature less than 212 degrees F and any control rod withdrawn from a core cell containing one or more fuel assemblies. The licensee proposes to delete the RPS LCOs, since it is only applicable to an operating reactor and does not apply with the reactor permanently shut down and defueled at VY.

Based on its review of the proposed change, the staff concludes that TS Section 3/4.1 is only needed to provide the LCOs, actions, and SRs necessary to maintain the ability of the RPS to automatically initiate a reactor scram to preserve the integrity of the fuel cladding, preserve the integrity of the RCPB, and minimize the energy, which must be absorbed following a LOCA. The RPS TS is only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the RPS system is no longer required.

The NRC staff also reviewed TS Section 3/4.1, which the licensee proposed for deletion to ensure that these LCOs are no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this SE. The staff notes that this TS indicates MODES for which the TS is applicable. MODES, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer applicable and is no longer required. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the staff finds that the proposed deletions of TS Section 3/4.1 are acceptable.

3.3.5 <u>TS Section 3/4.2, "Protective Instrument Systems"</u>

Section 3/4.2 of the VY TSs, "Protective Instrument Systems," contains the LCOs, actions, and SRs that provide for appropriate functional capability of sensing and control instrumentation required for safe operation of the facility. This section contains the following LCOs:

- Emergency Core Cooling System (ECCS)
- Primary Containment Isolation
- Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation
- Control Rod Block Actuation
- Mechanical Vacuum Pump Isolation Instrumentation
- Post-Accident Monitoring Instrumentation
- Recirculation Pump Trip Instrumentation
- Degraded Grid Protective System
- Reactor Core Isolation Cooling (RCIC) System Actuation

The ECCS TS specifies the operability requirements for ECCS protective instrumentation. This TS requires the ECCS to be operable in Run, Startup/Hot Standby, Hot Shutdown, Refuel with the reactor coolant temperature greater than 212 degrees F, and in Refuel with reactor coolant steam pressure greater than 150 psig.

The primary containment isolation TS specifies the operability requirements for the primary containment isolation instrumentation, which automatically initiates closure of appropriate primary containment isolation valves (PCIVs). This is required to be operable in Run, Startup/Hot Standby, Hot Shutdown, Refuel with the reactor coolant temperature greater than 212 degrees F, and in Refuel with any turbine stop valve or turbine bypass valve not closed.

The reactor building ventilation isolation and SGT system initiation TS specifies the operability requirements for the reactor building ventilation isolation and SGT system initiation instrumentation. The instrumentation automatically initiates closure of the reactor building automatic ventilation system isolation valves and starts the SGT system following DBAs requiring isolation of secondary containment. This reactor building ventilation isolation and SGT system initiation are required to be operable in Run, Startup/Hot Standby, Hot Shutdown, Refuel with the reactor coolant temperature greater than 212 degrees F, during operations with the potential for draining the reactor vessel, during movement of irradiated fuel assemblies or fuel cask in secondary containment, and during alteration of the reactor core.

The control rod block actuation TS specifies the operability requirements for the control rod block instrumentation. This control rod block actuation is required to be operable when rated thermal power is greater than 30 percent.

The mechanical vacuum pump isolation instrumentation TS specifies the operability requirements for the mechanical vacuum pump isolation instrumentation. This instrumentation initiates an isolation of the mechanical vacuum pump following events in which main steam radiation monitors exceed a predetermined value. Tripping and isolating the mechanical vacuum pumps limits control room and offsite doses in the event of a control rod drop accident. This mechanical vacuum pump isolation instrumentation is required to be operable in Run and Startup/Hot Standby when the mechanical vacuum pump is in service.

The post-accident monitoring instrumentation TS specifies the operability requirements for the post-accident monitoring instrumentation. The purpose of this instrumentation is to display in the control room, plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs. This post-accident monitoring instrumentation is required to be operable in Run and Startup/Hot Standby.

The recirculation pump trip (RPT) instrumentation TS specifies the operability requirements for the RPT instrumentation. The anticipated transient without scram (ATWS) prevention/mitigation system initiates a RPT, adding negative reactivity, following events in which a scram does not but should occur, to lessen the effects of an ATWS event. This RPT instrumentation is required to be operable in Run.

The degraded grid protective system TS specifies the operability requirements for the degraded grid protective system instrumentation. This instrumentation monitors the 4.16 kilovolt (kV) emergency buses. If the monitors determine that insufficient voltage is available and an ECCS initiation signal is present, the buses are disconnected from the offsite power sources and connected to the onsite emergency diesel generators (DG) power sources. This degraded grid protective system is required to be operable when the associated DG is required to be operable.

The RCIC system actuation TS specifies the operability requirements for the RCIC system instrumentation. This RCIC system actuation is required to be operable in Run, Startup/Hot Standby, Hot Shutdown, and in Refuel with reactor steam pressure greater than 150 psig.

The licensee proposes to delete all of the above TS LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

The NRC staff reviewed the proposed change to the ECCS Instrumentation and concludes that this TS provides the LCO and SRs necessary to maintain the instrumentation associated with the ECCS systems that provide cooling to the reactor core. Specifically, these systems are the core spray (CS) system, low pressure coolant injection (LPCI) system, high pressure coolant injection (HPCI) system, and automatic depressurization system (ADS), which provide adequate emergency cooling capability to the reactor in the event of loss of cooling to the reactor core while operating the plant. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel. Therefore, the ECCS instrumentation, which only addresses ECCSs, is no longer required because the accidents these systems and components were designed to mitigate are no longer possible. Based on the above, the staff finds that the proposed deletion of the TSs related to ECCS instrumentation is acceptable.

The NRC staff reviewed the proposed changes to the primary containment isolation, reactor building ventilation isolation, and SGT system initiation instrumentation. The staff concludes that these TSs provide the LCOs and SRs necessary to maintain the instrumentation associated with the PCIVs, the reactor building ventilation isolation, and SGT initiation. The PCIV instrumentation automatically initiates closure of appropriate PCIVs. The reactor building ventilation isolation and SGT system initiation automatically initiates closure of the reactor building automatic ventilation system isolation valves and starts the SGT system following DBAs requiring isolation of secondary containment. This instrumentation limits fission product release during and following postulated DBAs. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel at VY. Therefore, the PCIV instrumentation, the reactor building ventilation isolation, and SGT system initiation instrumentation supports primary and secondary containment isolation, are no longer required. Furthermore, the accidents these instruments were designed to contain are no longer possible. Therefore, none of the instrumentation addressed by these TSs are needed by a reactor that has permanently shut down and defueled in accordance with 10 CFR 50.82(a)(2).

The NRC staff reviewed the proposed change to the control rod block actuation instrumentation. The staff concludes that this TS provides the LCO and SRs necessary to maintain the instrumentation associated with control rod blocks. Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to backup administrative controls on control rod movement. During shutdown conditions, control rod blocks from the Reactor Mode Switch-

Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the control rod block actuation instrumentation is no longer required.

The NRC staff reviewed the proposed change to the mechanical vacuum pump isolation instrumentation. The staff concludes that this TS provides the LCO and SRs necessary to maintain the instrumentation associated with the mechanical vacuum pump isolation. This instrumentation initiates an isolation of the mechanical vacuum pump following events in which main steam radiation monitors exceed a predetermined value. Tripping and isolating the mechanical vacuum pumps limits control room and offsite doses in the event of a control rod drop accident. The control rod drop accident instrumentation was designed to protect against control rod drop accident, is no longer possible because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the mechanical vacuum pump isolation instrumentation is no longer required.

The NRC staff reviewed the proposed change to the post-accident monitoring instrumentation. The staff concludes that this TS provides the LCO and SRs necessary to ensure there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident and allows operators to take manual actions specified in the emergency operating procedures. The purpose of this instrumentation is to display, in the control room, plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs. The DBAs that this instrumentation was designed to monitor are no longer possible because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the post-accident monitoring instrumentation is no longer required for plant safety.

The NRC staff reviewed the proposed change to the RPT instrumentation. The staff concludes that this TS provides the LCO and SRs necessary to maintain the instrumentation associated with the ATWS Prevention/Mitigation System. The ATWS Prevention/Mitigation System initiates a RPT, adding negative reactivity, following events in which a scram does not but should occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. The ATWS event that this instrumentation was designed to protect against is no longer possible because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the RPT instrumentation is no longer required for plant safety.

The NRC staff reviewed the proposed change to the degraded grid protective system instrumentation. The staff concludes that this TS provides the LCO and SRs necessary to ensure the capability to detect a loss of offsite power and automatically actuate DGs to provide safety related power for SSCs needed for accident mitigation. The degraded grid protective system instrumentation monitors power available to the 4.16 kV emergency buses. Offsite power is the preferred source of power for the 4.16 kV emergency buses. If the degraded grid protective system instrumentation determines that insufficient voltage is available to the 4.16 kV emergency buses and an ECCS initiation signal is present, the buses are disconnected from the offsite power sources and connected to the onsite DGs. The DBAs that this instrumentation was designed to mitigate are no longer possible because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the degraded grid protective system instrumentation is no longer required for plant safety.

The NRC staff reviewed the proposed change to the RCIC system actuation instrumentation. The staff concludes that this TS provides the LCO and SRs necessary to maintain the instrumentation associated with RCIC. The purpose of the RCIC system instrumentation is to initiate action to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (main condenser) and normal coolant makeup flow from the reactor feedwater system is insufficient or unavailable, such that RCIC System initiation occurs and maintains sufficient reactor water level such that initiation of the low pressure ECCS pumps does not occur. The DBAs that this instrumentation was designed to mitigate are no longer possible because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the RCIC system instrumentation is no longer required for plant safety.

The NRC staff also reviewed TS Section 3/4.2, "Protective Instrument Systems," which the licensee proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer required. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the staff finds that the proposed deletions of TS Section 3/4.2 are acceptable.

3.3.6 TS Section 3/4.3, "Control Rod System"

Section 3/4.3 of the VY TSs, "Control Rod Systems," contains the LCOs, actions, and SRs that provide the ability of the control rods to control reactivity in the reactor and, in turn, protect the integrity of a fission product barrier. This section contains the following LCOs:

- Reactivity Limitations
- Control Rods
- Scram Insertion Times
- Control Rod Accumulators
- Reactivity Anomalies
- Scram Discharge Volume Vent and Drain Valves

The licensee proposes to delete all of the above TSs LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.3.A, "Reactivity Limitations," ensures that the core loading is limited to that which can be made subcritical in the most reactive condition during the operation cycle, with the highest worth operable control rod in its fully withdrawn position and all other operable rods inserted. This TS applies only when fuel is in the reactor vessel core.

TS 3/4.3.B, "Control Rods," establishes conditions to prevent the addition of excess reactivity due to improper operation of the control rods or failure of the control rod system. This TS applies in all modes unless specified.

TS 3/4.3.C, "Scram Insertion Times," establishes conditions to ensure that the control rods would insert to bring the reactor subcritical at a rate fast enough to prevent fuel damage during limiting power transients. These TSs apply when the reactor is in Run and Startup modes.

TS 3/4.3.D, "Control Rod Accumulators," establishes conditions to limit the number and location of control rod accumulators allowed to be inoperable to ensure that the control rods scram under varying reactor conditions that the reactor could be made subcritical. This TS applies when the reactor is in Run and Startup modes.

TS 3/4.3.E, "Reactivity Anomalies," establishes requirements to shut down the reactor if the difference between the actual critical rod configuration and the expected configuration during power operation exceeds 1 percent delta K per K. This TS applies when the reactor is in Run mode.

TS 3/4.3.F, "Scram Discharge Volume Vent and Drain Valves," ensures that sufficient volume is available at all times to allow a complete scram of all the control rods and thus bring the reactor subcritical. This TS applies when the reactor is in Run and Startup modes.

The NRC staff reviewed these proposed changes. The staff concludes that these TSs are only needed to provide the LCOs, actions, and SRs necessary to maintain reactivity parameters of the fuel loaded into a reactor vessel within the margins of conditions encountered during normal operation, anticipated occurrences, and for DBAs. The control rod system TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the control rod systems are no longer required.

The NRC staff also reviewed TS Section 3/4.3, "Control Rod System," proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer required. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the staff finds that the proposed deletions of TS Section 3/4.3 are acceptable.

3.3.7 TS Section 3/4.4, "Reactor Standby Liquid Control System"

Section 3/4.4 of the VY TSs, "Reactor Standby Liquid Control System," contains the LCOs, actions, and SRs that provide assurance of availability and operability of the standby liquid control system as a means of providing an independent reactivity control mechanism. This section contains the following LCOs:

- Normal Operation
- Operation with Inoperable Components
- Standby Liquid Control System Tank - Borated Solution

The design objective of the Reactor Standby Liquid Control System is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. These TSs are required when the reactor is in Run and Startup/Hot Standby modes.

The reactor standby liquid control system TSs provide the requirements needed to meet the design objective and the requirements of the ATWS rule, 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." The licensee proposes to delete all of the above LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

The NRC staff reviewed the proposed change and concludes that because (1) 10 CFR 50.62 does not apply to nuclear power reactor facilities for which the certifications required under 10 CFR 50.82(a)(1) have been submitted and (2) 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the reactor standby liquid control system is no longer required. The reactor standby liquid control system TSs are only applicable to a reactor authorized to operate or retain irradiated fuel in the reactor vessel.

The NRC staff also reviewed the Section 3/4.4, "Reactor Standby Liquid Control System," TSs proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer applicable and is no longer required. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply. Based on the above, the deletion of the Section 3/4.4, "Reactor Standby Liquid Control System" TSs are acceptable.

3.3.8 <u>Section 3/4.5, "Core and Containment Cooling Systems"</u>

Section 3/4.5 of the VY TSs, "Core and Containment Cooling Systems," contains the LCOs, actions, and SRs that provide assurance of adequate cooling capability for heat removal in the event of a LOCA or isolation from the normal reactor heat sink, the main condenser. This section contains the following LCOs:

- Core Spray and Low Pressure Coolant Injection
- Containment Spray Cooling Capability
- Residual Heat Removal (RHR) Service Water System
- Station Service Water and Alternate Cooling Tower Systems
- High Pressure Cooling Injection (HPCI) System
- Automatic Depressurization System
- Reactor Core Isolation Cooling System (RCIC)
- Minimum Core and Containment Cooling System Availability
- Maintenance of Filled Discharge Pipe

The licensee proposes to delete all of the above TSs LCOs since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.5.A, "Core Spray and Low Pressure Coolant Injection," provides sufficient cooling to the core to dissipate the energy associated with a LOCA. This TS applies whenever irradiated fuel is in the reactor vessel and prior to a reactor startup from a cold shutdown condition.

TS 3/4.5.B, "Containment Spray Cooling Capability," and TS 3/4.5.C, "Residual Heat Removal (RHR) Service Water System," ensure the ability to remove heat energy from the containment in the event of a LOCA. These TSs require both containment cooling spray loops be operable when the reactor water temperature is greater than 212 degrees F.

TS 3/4.5.C, ensures the ability to remove heat energy from the containment in the event of a LOCA. This TS applies whenever irradiated fuel is in the reactor vessel and prior to a reactor startup from a cold shutdown condition.

TS 3/4.5.D, "Station Service Water and Alternate Cooling Tower System." The station service water system essential equipment cooling loops provide redundant heat sinks to dissipate residual heat in the reactor fuel after a shutdown or accident. The alternate cooling tower system provides the necessary heat sink for normal post-shutdown conditions in the event that the station service water system becomes incapacitated. This TS applies whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212 degrees F.

TS 3/4.5.E, "High Pressure Cooling Injection (HPCI) System," provides adequate cooling to the core for all pipe breaks smaller than those for which the LPCI or CS subsystems can protect the core. This TS applies whenever irradiated fuel is in the reactor vessel and the reactor steam pressure is greater than 150 psig.

TS 3/4.5.F, "Automatic Depressurization System," is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain sufficient reactor water level. This TS applies whenever irradiated fuel is in the reactor vessel and the reactor steam pressure is greater than 150 psig.

TS 3/4.5.G, "Reactor Core Isolation Cooling System (RCIC)," is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the ECCS. This TS applies whenever irradiated fuel is in the reactor vessel and the reactor steam pressure is greater than 150 psig.

TS 3/4.5.H, "Minimum Core and Containment Cooling System Availability," provides a method of transferring the residual heat from the reactor vessel following a shutdown or accident to a heat sink. This TS applies when irradiated fuel is in the reactor vessel and the reactor is in either a refueling of cold shutdown condition.

TS 3/4.5.I, "Maintenance of Filled Discharge Pipe," requires maintaining the discharge lines of the CS subsystems, LPCI subsystems, HPCI system, and RCIC system full of water to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start. This TS applies whenever the CS subsystems, LPCI subsystem, HPCI or RCIC are required to be operable.

The NRC staff reviewed the proposed changes to the core and containment cooling systems' TSs and has determined that these TSs are only needed to provide the LCOs, actions, and SRs necessary to maintain functionality of the systems that provide cooling to the reactor core in the event of a LOCA or isolation from the normal reactor heat sink. Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the Commission that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this condition, a LOCA will no longer be possible. Based on this, the staff concludes the core and containment cooling systems are no longer required at VY.

The NRC staff also reviewed TS Section 3/4.5, "Core And Containment Cooling Systems," which the licensee proposed for deletion, to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer applicable and is no longer required. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

The regulation in 10 CFR 50.46(a)(1)(i) requires each boiling light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an ECCS designed to meet the cooling performance criteria following postulated LOCAs. Thus, 10 CFR 50.46 does not apply to nuclear power reactor facilities for which the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Based on the above, the NRC staff finds that the proposed deletions of TS Section 3/4.5 are acceptable.

3.3.9 Section 3/4.6, "Reactor Coolant System"

Section 3/4.6 of the VY TSs, "Reactor Coolant System," contains the LCOs, actions, and SRs that provide assurance of the integrity and safe operation of the RCS. This section contains the following LCOs:

- Pressure and Temperature Limitations
- Coolant Chemistry
- Coolant Leakage
- Safety and Relief Valves
- Structural Integrity and Operability Testing
- Jet Pumps
- Single Loop Operation
- Recirculation System
- Thermal Hydraulic Stability

The licensee proposes to delete all of the above TS LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.6.A, "Pressure and Temperature Limitations," specifies the maximum specified heatup and cooldown rates during reactor startup and shutdown to be consistent with the design assumptions and satisfy the stress limits for cyclic operation. This TS applies in all modes of normal operation.

TS 3/4.6.B, "Coolant Chemistry," specifies the radiochemistry limits for the reactor coolant in the RCS, which limit potential offsite doses due to postulated events. This TS applies for reactor power operation.

TS 3/4.6.C, "Coolant Leakage," specifies the maximum leakage rates of reactor coolant from the RCS, which allows prompt identification and isolation of leaks before the integrity of the RCS pressure boundary is impaired. This TS applies any time irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212 degrees F.

TS 3/4.6.D, "Safety and Relief Valves," specifies the requirements for the safety and relief valves, which are designed to prevent overpressurization of, and damage to, the RCS boundary. This TS applies during reactor power operation and whenever the reactor coolant pressure is greater than 150 psig and the reactor coolant temperature is greater than 350 degrees F.

TS 3/4.6.E, "Structural Integrity and Operability Testing," specifies the requirements for structural integrity and operability testing of safety related components, which addresses the inservice inspection requirements of the RCS boundary components. This TS applies in all modes of normal operation.

TS 3/4.6.F, "Jet Pumps," specifies the requirements for the jet pumps. The jet pumps function as part of the reactor recirculation system to provide sufficient coolant circulation through the core during normal power operation to maintain normal operating temperatures. This TS applies when the reactor is in Run and Startup/Hot Standby modes.

TS 3/4.6.G, "Single Loop Operation," specifies the system settings and configuration necessary to operate the reactor with one recirculation loop, which prevents a cold water injection transient caused by an inadvertent pump start. This TS applies when the reactor is in Run and Startup/Hot Standby modes.

TS 3/4.6.H, "Recirculation System," specifies the requirements to shutdown the reactor when the recirculation system is not in operation. The reactor recirculation system consists of the two recirculation pump loops external to the reactor vessel, which provide the driving flow of water to the reactor vessel jet pumps. The reactor recirculation system provides sufficient subcooled water to the core during normal power operation to maintain normal operating temperatures. This TS applies in all modes of normal operation.

TS 3/4.6.J, "Thermal Hydraulic Stability," specifies restrictions against operating the reactor in the power flow exclusion area. Reactor coolant flow instabilities are likely if the reactor is operated in the power flow exclusion area, and if not prevented or suppressed, can cause exceeding the specified fuel design limits. This TS applies when the reactor is in Run mode.

The NRC staff reviewed these proposed changes and concludes that these TSs provide the LCOs and SRs necessary to maintain the integrity of the RCS pressure boundary. All of these TSs are related to assuring the integrity of the RCS pressure boundary. Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the Commission that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. Therefore, the TSs contained in Section 3/4.6, which only address the RCPB, are no longer applicable.

The NRC staff also reviewed Section 3/4.6, "Reactor Coolant System," which the licensee proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in the TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer applicable and is no longer required. Because ENO previously submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the staff finds that the proposed deletions of TS Section 3/4.6 are acceptable.

3.3.10 Section 3/4.7, "Station Containment Systems"

Section 3/4.7 of the VY TSs, "Station Containment Systems," contains the LCOs, actions, and SRs that assure the integrity of the primary and secondary containment systems. This section contains the following LCOs:

- Primary Containment
- Standby Gas Treatment System
- Secondary Containment System
- Primary Containment Isolation Valves
- Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

The licensee proposes to delete all of the above TSs LCOs since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.7.A, "Primary Containment," specifies the operability requirements for primary containment. The primary containment consists of a drywell, which encloses the reactor vessel and recirculation system, a pressure suppression chamber, which stores a large volume of water, a connecting vent system between the drywell and the suppression chamber, isolation valves, containment cooling systems, and other service equipment. The safety objective of the primary containment system in conjunction with the core standby cooling systems is to provide the capability, in the event of a postulated LOCA, to limit the release of fission products to the plant environment so that offsite doses remain well below the values specified in 10 CFR 50.67. These TSs are required when the reactor is critical or when the reactor water temperature is above 212 degrees F and fuel is in the reactor vessel.

TS 3/4.7.B, "Standby Gas Treatment System," specifies the operability requirements for the Standby Gas Treatment (SGT) system. The SGT system is provided to limit airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the reactor building. This TS applies whenever the reactor is in Run and Startup/Hot Standby modes.

TS 3/4.7.C, "Secondary Containment System," specifies the operability requirements for secondary containment. The secondary containment system consists of the reactor building, the SGT system, the reactor building normal heating ventilating and air conditioning system, and the reactor building penetrations. The secondary containment is designed to minimize the ground level release of airborne radioactive material and to control this release should an accident occur. This TS applies whenever the reactor is in Run and Startup/Hot Standby modes, during movement of irradiated fuel assemblies or the fuel cask in secondary containment, during alteration of the reactor core, or during operation with the potential for draining the reactor vessel.

TS 3/4.7.D, "Primary Containment Isolation Valves," specifies the operability requirements for PCIV. The primary containment design includes lines that penetrate the primary containment with different containment isolation valve configurations including double and single valve isolation. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA. This TS applies when the reactor is in Run mode.

TS 3/4.7.E, "Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIV)," specifies the operability requirements for the RBAVSIVs when secondary containment integrity is required. The function of the RBAVSIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. This TS applies whenever the reactor is in Run and Startup/Hot Standby modes, during movement of irradiated fuel assemblies or the fuel cask in secondary containment, during alteration of the reactor core, or during operation with the potential for draining the reactor vessel.

The NRC staff reviewed the proposed changes and concludes that these TSs provide the LCOs and SRs necessary to maintain the integrity of the primary and secondary containment systems. Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the Commission that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this condition, the operational conditions, transients, and accidents the containment SSCs were designed to contain are no longer possible, and there will be no challenge to containment integrity. The only DBA that could potentially apply at this time to the permanently shutdown and defueled VY reactor would be the FHA.

The licensee proposed a revision to the FHA in an application dated November 14, 2013 (ADAMS Accession No. ML13323A518), "Eliminate Operability Requirements for Secondary Containment When Handling Sufficiently Decayed Irradiated Fuel or Fuel Cask." The FHA analysis does not rely on primary containment or the primary containment isolation valves to mitigate the consequences of an FHA and after 13 days of fuel decay, secondary containment, RBAVSIVs, and SGT are no longer required to mitigate the consequences of the FHA. In November 14, 2013, application, the licensee stated that the changes proposed by this amendment would not be effective until 13 days after final reactor shutdown to allow for sufficient decay of the reactor fuel such that the fuel will no longer be considered recently irradiated, as defined in the FHA. The NRC staff's confirmatory analyses of VY's revised FHA yields results of the EAB, LPZ and CR doses that are less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria.

Therefore, the TSs contained in Section 3/4.7, which address primary and secondary containment integrity, isolation, and leakage are no longer applicable following 13 days of fuel decay time after reactor shutdown. Furthermore, the accidents these systems and components were designed to contain are no longer possible, with the exception of the FHA. In the case of the FHA, the licensee stated that November 14, 2013, licensee amendment would not be implemented until 13 days after final reactor shutdown, which occurred on December 29, 2014. Therefore, based on NRC approval of the License Amendment No. 262, dated February 12, 2015 (ADAMS Accession No. ML14304A588), "Issuance of Amendment - Eliminate Operability Requirements for Secondary Containment When Handling Sufficiently Decayed Irradiated Fuel or Fuel Cask," and as stated above, the proposed deletion of Section 3/4.7, "Station Containment Systems" TSs are acceptable.

3.3.11 Section 3/4.8, "Radioactive Effluents"

Section 3/4.8 of the VY TSs, "Radioactive Effluents," contains the LCOs, actions, and SRs that assure that radioactive effluents are kept "as low as is reasonably achievable" in accordance with 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low as is Reasonably Achievable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," and, in any event, are within the dose limits for members of the public as specified in 10 CFR 20, "Standards for Protection Against Radiation," and applies to the release of all radioactive effluents from the plant. This section contains the following LCOs:

- Liquid Holdup Tanks
- Explosive Gas Mixture
- Steam Jet Air Ejector (SJAE)

The licensee proposes to renumber the section title, correct a typographical error in the title of current TS 3/4.8 and delete the "Explosive Gas Mixture" and "Steam Jet Air Ejector (SJAE)" TS LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.8.D, "Liquid Holdup Tanks," specifies the allowed quantities of radioactive material in the liquid holdup tanks. The licensee is not proposing any changes to this LCO.

TS 3/4.8.J, "Explosive Gas Mixture," specifies the limits and monitoring requirements for hydrogen concentration in the offgas downstream of the operating recombiner in the advanced offgas system. The hydrogen monitors are used to detect possible hydrogen buildups, which could result in a possible hydrogen explosion. This TS applies in all modes of normal operation.

TS 3/4.8.K, "Steam Jet Air Ejector (SJAE)," specifies limits on the gross radioactivity release rate of gases from the SJAE to provide reasonable assurance that the TEDE to an individual at the EAB will not exceed the limits of 10 CFR 50.67 in the event this effluent is inadvertently discharged directly to the environment without treatment. SJAEs are used to remove gases from the main condenser during normal reactor operation. This TS applies in all modes of normal operation.

Based on its review of the proposed changes, the NRC staff concludes that these TSs provide the LCOs, actions, and SRs necessary to that assure that radioactive effluents are kept as low as is reasonably achievable in accordance with 10 CFR 50, Appendix I, and are within the dose limits for members of the public specified in 10 CFR 20 when fuel is in reactor. Once the reactor is in the permanently defueled condition, hydrogen will no longer be produced by reactor operation and explosive gas mixture monitoring will no longer be required. The explosive gas mixture and SJAE are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core and the explosive gas mixture and SJAE systems are no longer required.

The NRC staff also reviewed TS Section 3/4.8, "Explosive Gas Mixture," and "Steam Jet Air Ejector (SJAE)," which the licensee proposed for deletion, to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in the TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no required for reactor safety. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the staff finds that the proposed deletions of TS Section 3/4.8, "Explosive Gas Mixture" and "Steam Jet Air Ejector (SJAE)" are acceptable.

3.3.12 <u>Section 3/4.10, "Auxiliary Electrical Power Systems"</u>

Section 3/4.10 of the VY TSs, "Auxiliary Electrical Power Systems," contains the LCOs, actions, and SRs that assure an adequate supply of electrical power for operation of those systems required for reactor safety. This section contains the following LCOs:

- Normal Operation
- Operation with Inoperable Components
- Diesel Fuel

The licensee proposes to delete all of the above TSs LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.10.A, "Normal Operation," specifies requirements to ensure that the offsite power sources, the onsite standby power sources, emergency buses and RPS power protection provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to engineered safety features systems during and following DBAs so that the fuel, RCS, and containment design limits are not exceeded. This TS applies in Run and Startup/Hot Standby modes.

TS 3/4.10.B, "Operation With Inoperable Components," specifies requirements that ensure adequate power is available to operate the emergency safeguards equipment from the immediate access source or for minimum engineered safety features from either of the emergency DGs. This TS applies in Run and Startup modes with the reactor not in the cold condition.

TS 3/4.10.C, "Diesel Fuel," specifies the requirements for stored diesel fuel oil. Stored diesel fuel oil is required to have sufficient supply for 7 days of rated load operation for a single emergency DG. It also ensures the stored diesel fuel oil meets the quality standards. This TS applies in all modes of normal operation.

The NRC staff reviewed the proposed changes and concludes that these TSs provide the LCOs, actions, and SRs necessary to assure an adequate supply of electrical power for operation of those systems required for reactor safety. Under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized by the 10 CFR Part 50 license when the licensee submits a certification to the Commission that power operations have been permanently ceased and that the fuel has been permanently removed from the reactor vessel. In this condition, the operational conditions, transients, and postulated DBAs are no longer possible. Therefore, the systems required for reactor safety that the auxiliary electrical power systems were designed to power are no longer needed. The only DBA that could currently apply to the permanently shutdown and defueled VY reactor would be the FHA.

The licensee proposed a revision to the FHA in a letter dated November 14, 2013. (ADAMS Accession No. ML13323A518). The FHA analysis does not rely on normal or emergency power to mitigate the consequences of an FHA and alternating current and direct current sources are no longer required during movement of irradiated fuel assemblies for mitigation of a potential FHA. In the application of November 14, 2013, the licensee stated that the changes proposed by this amendment would not be effective until 13 days after final reactor shutdown, which occurred on December 29, 2014, to allow for sufficient decay of the reactor fuel such that the fuel will no longer be considered recently irradiated, as defined in the FHA.

Therefore, the TSs contained in Section 3/4.10, which address auxiliary electrical power systems are no longer applicable. Furthermore, the accidents these systems and components were designed to mitigate are no longer possible, with the exception of the FHA. In the case of the FHA, the licensee stated that this licensee amendment would not be implemented until 13 days after final reactor shutdown. Therefore, based on NRC approval the License Amendment No. 262, dated February 12, 2015, "Issuance of Amendment - Eliminate Operability Requirements for Secondary Containment When Handling Sufficiently Decayed Irradiated Fuel or Fuel Cask," and that stated above, the proposed deletion of TSs related to auxiliary electrical power systems are acceptable.

The NRC staff also reviewed Section 3/4.10, "Auxiliary Electrical Power Systems," TSs proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in the TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer required for reactor safety. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the NRC staff finds that the proposed deletions of TS Section 3/4.10 are acceptable.

3.3.13 <u>Section 3/4.11, "Reactor Fuel Assemblies"</u>

Section 3/4.11 of the VY TSs, "Reactor Fuel Assemblies," contains the LCOs, actions, and SRs that assure the performance of the fuel rods. This section contains the following LCOs:

- Average Planar Linear Heat Generation Rate (APLHGR)
- Linear Heat Generation Rate (LHGR)
- Minimum Critical Power Ratio (MCPR)

The licensee proposes to delete all of the above TSs LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.11.A, "Average Planar Linear Heat Generation Rate (APLHGR)," specifies limits to ensure that the peak cladding temperature during a design-basis LOCA does not exceed 2200 degrees F. This TS applies in Run mode when greater or equal to 23 percent rated thermal power.

TS 3/4.11.B, "Linear Heat Generation Rate (LHGR)," specifies requirements for the LHGR of a fuel rod at a given nodal plane in a bundle. These limits are bundle type dependent and are monitored to assure all mechanical design requirements are met. This TS applies in Run mode when greater or equal to 23 percent rated thermal power.

TS 3/4.11.C, "Minimum Critical Power Ratio (MCPR)," specifies the MCPR operating limit, which is a cycle-dependent parameter that can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provide reasonable assurance against exceeding the fuel cladding integrity safety limit for potential abnormal occurrences. This TS applies in Run mode when greater or equal to 23 percent rated thermal power.

The NRC staff reviewed the proposed deletions and concludes that these TSs are only needed to provide the LCOs and SRs necessary to maintain reactor power and heat generation within the margins of conditions encountered during normal operation, anticipated occurrences, and for DBAs. The reactor fuel assemblies TSs are only important for a reactor authorized to operate. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, there is no reactor core generating power and the reactor fuel assemblies limits are no longer required for reactor safety.

The NRC staff has also reviewed Section 3/4.11 TSs proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in the TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES for a permanently shutdown and defueled reactor, such as VY, is no longer required for reactor safety. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the NRC staff finds that the proposed deletions of TS Section 3/4.11 are acceptable.

3.3.14 Section 3/4.12, "Refueling and Spent Fuel Handling"

Section 3/4.12 of the VY TSs, "Refueling and Spent Fuel Handling," contains the LCOs, actions, and SRs that assure core reactivity is within capability of the control rods, to prevent criticality during refueling, and to assure safe handling of spent fuel casks. This section contains the following LCOs:

- Refueling Interlocks
- Core Monitoring
- Fuel Storage Pool Water Level
- Control Rod and Control Rod Drive Maintenance
- Extended Core Maintenance
- Fuel Movement
- Spent Fuel Pool Water Temperature

The licensee has proposed to:

1. Delete the Refueling Interlocks, Core Monitoring, Control Rod and Control Rod Drive Maintenance, Extended Core Maintenance, and Fuel Movement LCOs.
2. Revise the title of the Section 3/4.12.

3. Revise the applicability and objective for Section 3/4.12.
4. Revise 4.12.H Spent Fuel Pool Water Temperature.
5. Renumber Section 3/4.12.

3.3.14.1 Deletion of TSs: "Refueling Interlocks," "Core Monitoring," "Control Rod and Control Rod Drive Maintenance," "Extended Core Maintenance," and "Fuel Movement," LCOs

The licensee proposes to delete the "Refueling Interlocks," "Core Monitoring," "Control Rod and Control Rod Drive Maintenance," "Extended Core Maintenance," and "Fuel Movement," TSs LCOs since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

TS 3/4.12.A, "Refueling Interlocks," specifies requirements for refueling interlocks and restriction of certain refueling procedures, such that, there is assurance that inadvertent criticality does not occur during refueling. This TS applies during in vessel fuel movements.

TS 3/4.12.B, "Core Monitoring," monitors the core reactivity condition during periods of station shutdown and refueling operations and guides the operator during refueling operations and station startup. This TS applies during core alterations.

TS 3/4.12.D, "Control Rod and Control Rod Drive Maintenance," specifies requirements for performing control rod/control rod drive maintenance and maintaining adequate shutdown margin during such activities to provide assurance that inadvertent criticality does not occur during such maintenance. This TS applies during in vessel fuel movements.

TS 3/4.12.E, "Extended Core Maintenance," specifies requirements for unloading of a portion of the reactor core for such purposes as inservice inspection requirements, examination of the core support plate, control rod, control rod drive maintenance, and provides assurance that inadvertent criticality does not occur during such operation. This TS applies during invessel fuel movements.

TS 3/4.12.F, "Fuel Movement," specifies requirements to assure that the reactor core has been shut down for at least 24 hours following power operation and prior to fuel handing or movement. This TS applies in all modes of normal operation.

The NRC staff reviewed these proposed changes in TS 3/4.12.A, TS 3/4.12.B, TS 3/4.12.D, TS 3/4.12.E, and TS 3/4.12.F and concludes that these TS are only needed to provide the LCOs, actions, and SRs necessary to assure core reactivity is within capability of the control rods, and to prevent criticality during refueling operations. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, refueling is no longer allowed and these TSs are no longer required for reactor safety.

The NRC staff also reviewed TS Section 3/4.12, "Refueling Interlocks," "Core Monitoring," "Control Rod and Control Rod Drive Maintenance," "Extended Core Maintenance," and "Fuel Movement," which the licensee proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in the TSs, as described in Section 2.0 of this SE. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODE, as defined in the TSs, is established by the mode-selector-switch. The reference to MODES is no longer required for a permanently shutdown and defueled reactor, such as VY. Because ENO has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and VY is no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the NRC staff finds that the proposed changes to TS Section 3/4.12 are acceptable.

3.3.14.2 Section 3/4.12 Revised Title

The current title of Section 3/4.12 is "Refueling and Spent Fuel Handling." The licensee is proposing to revise the title to "Spent Fuel Storage," so that it reflects that refueling of the reactor will no longer be possible.

The NRC staff reviewed this proposed change and concludes that because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, refueling is no longer allowed and that the proposed change accurately describes this condition and is consistent with 10 CFR 50.82 for VY.

3.3.14.3 Section 3/4.12 Revised Applicability and Objective

The current TS Section 3/4.12 Applicability and Objective are:

LCO 3.12 Applicability:

Applies to fuel handling, core reactivity limitations, and spent fuel handling.

LCO 3.12 Objective:

To assure core reactivity is within capability of the control rods, to prevent criticality during refueling, and to assure safe handling of spent fuel casks.

SR 4.12 Applicability:

Applies to periodic testing of those interlocks and instruments used during refueling.

SR 4.12 Objective:

To verify the operability of instrumentation and interlocks used in refueling.

The licensee proposes to revise and renumber the Applicability and Objective to:

LCO 3.2 Applicability:

Applies to storage of spent fuel.

LCO 3.2 Objective:

To assure safe storage of spent fuel.

SR 4.2 Applicability:

Applies to the parameters which monitor the storage of spent fuel.

SR 4.2 Objective:

To verify that spent fuel is being stored safely.

The NRC staff reviewed the proposed wording changes to Section 3/4.12 Applicability and Objective and concludes that they are consistent with the transition from an operating reactor to a permanently shutdown and defueled facility with a primary safety focus of safe storage of spent fuel. In addition, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, refueling is no longer allowed.

The NRC staff finds that the proposed changes stated above reflect the change in plant status and are appropriate. Therefore, the staff finds the proposed changes to be acceptable.

3.3.14.4 Revised SR 4.12.H, "Spent Fuel Pool Water Temperature"

TS 4.12.H currently states:

> 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.

The licensee proposes to remove the word "refueling" from TS 4.12.H, so that it will state:

> 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 operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.

Based on its review of the proposed changes, the NRC staff concludes that the change is consistent with the transition from an operating reactor to a permanently shutdown and defueled facility with a primary safety focus of safe storage of spent fuel. In addition, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at VY, refueling is no longer allowed. The staff finds that the proposed change reflects the change in plant status and is appropriate. Therefore, the proposed change is acceptable.

3.3.14.5 Section 3/4.12 Renumbering and Re-lettering

The licensee proposes to renumber TS 3/4.12 to TS 3/4.2 and to re-letter TS 3/4.12.C and TS 3/4.12.H to TS 3/4.2.A and TS 3/4.2.B. The NRC staff reviewed the proposed changes and concludes that they are of a clarifying nature and do not change any technical requirements; therefore, these changes are acceptable.

3.3.15 Section 5.0, "Design Features"

This section contains a description of the design features of the facility.

3.3.15.1 TS 5.1, "Site"

The licensee proposes to retain this section in the TSs without modification or revision.

3.3.15.2 TS 5.2, "Reactor," TS 5.3, "Reactor Vessel," and TS 5.4, "Containment"

These TSs provide a general description of the number of and design material requirements for the fuel, and control rod assemblies, in addition to the description of the reactor vessel, and the principal design parameters and applicable design codes for the reactor vessel, secondary, and primary containments. The licensee proposes to delete the requirements for fuel and control rod assemblies, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of VY.

The three TSs to be deleted address the reactor, reactor vessel, and containment. These structures and systems serve to prevent or mitigate the consequences of postulated accidents involving the release of fission products. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel. In addition, the accidents these systems and components were designed to mitigate are no longer possible. Therefore, these TSs are no longer applicable.

Based on the above, the NRC staff finds that the proposed deletions of the design features TS sections, titled "Reactor," "Reactor Vessel," and "Containment" are acceptable.

3.3.15.3 Spent and New Fuel Storage

This TS provides a general description of the new and spent fuel storage facilities.

The current title of TS Section 5.5 is "Spent and New Fuel Storage."

The licensee proposes to revise the title and renumber TS Section 5.5 to TS Section 5.2 "Spent Fuel Storage." The licensee also proposes to re-letter TS 5.5.B to TS 5.5.A; TS 5.5.C to TS 5.5.B; TS 5.5.D to TS 5.5.C; and TS 5.5.E to TS 5.5D.

Current TS 5.5.A states:

> The new fuel storage facility shall be such that the effective multiplication factor (k_{eff}) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.

The licensee proposes to delete the new fuel storage facility requirement, since the new fuel storage facility will not be used to store spent fuel and new fuel is no longer stored on site. In addition, the licensee proposes to delete the reference to the new fuel storage facility in current TS 5.5.E as shown by the ~~strikethrough~~ below, and requests that License Condition 2.B be revised to no longer allow receipt of new fuel.

> The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool ~~or the new fuel storage facility~~ shall be less than or equal to 1.31 at 20°C.

The NRC staff reviewed the proposed changes to the spent and new fuel storage requirements. The licensee has permanently shutdown and defueled the reactor, and will no longer be receiving any new fuel onsite because 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel at VY. In addition, as evaluated below, the licensee proposed that License Condition 2.B be revised to no longer allow receipt of new fuel. Since the facility license will no longer allow new fuel to be stored onsite, the requirements for the new fuel storage facility are no longer applicable.

Based on the above, the staff finds that the proposed change to delete the new fuel storage facility requirements is acceptable.

3.3.16 Section 6.0, "Administrative Controls"

This section contains a description of the administrative controls at the facility. The licensee proposed revision to or deletion of the administrative controls in a letter dated October 31, 2013 (ADAMS Accession Number ML13316A004). In this application the licensee proposed to remove portions of the TSs that are no longer applicable to the facility in its permanently defueled condition. Subsequently, the NRC approved the changes in Section 6.0 on December 22, 2014, by License Amendment No. 260 to Renewed Facility Operating License No. DPR-28 (ADAMS Accession No. ML14217A072).

Based on NRC approval of the requested changes by License Amendment No. 260 stated above, the proposed changes in this application are of a clarifying nature and do not change any technical requirements; therefore, these changes are acceptable. However, the current page numbers 255 through 267 in Section 6.0 will be renumbered to the new page numbers 12 through 20 as a result of this amendment.

3.4 <u>Changes to Renewed Facility Operating License</u>

3.4.1 Changes to License Condition a.

Current License Condition a states:

a. Construction of the Vermont Yankee Nuclear Power Station (the facility) has been substantially completed in conformity with the application, as amended, the Provisional Construction Permit No. CPPR-36, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in Title 10 of the *Code of Federal Regulations* (CFR) Chapter I: and

Revised License Condition a would state:

a. This paragraph deleted by Amendment No. 263.

ENO proposed to delete License Condition a. because it has no bearing on the regulation of decommissioning activities at VY.

3.4.2 Changes to License Conditions b, c, d, f, g, and h

Current License Condition b states:

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

Revised License Condition b would state:

b. The facility is prohibited from operating the reactor in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and

Current License Condition c states:

c. There is reasonable assurance (1) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and

Revised License Condition c. would state:

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

Current License Condition d states:

d. Entergy Nuclear Vermont Yankee, LLC is financially qualified and Entergy Nuclear Operations, Inc. is technically and financially qualified to engage in the activities authorized by this renewed operating license, in accordance with the rules and regulations of the Commission; and

Revised License Condition d would state:

d. Entergy Nuclear Vermont Yankee, LLC is financially qualified and Entergy Nuclear Operations, Inc. is technically and financially qualified to engage in the activities authorized by this license, in accordance with the rules and regulations of the Commission; and

Current License Condition f states:

f. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public; and

Revised License Condition f would state:

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

Current License Condition g states:

g. After weighing the environmental, economic, technical and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this renewed operating license (subject to the conditions for protection of the environment set forth herein) is in accordance with 10 CFR Part 51, of the Commission's regulations and all applicable requirements of said Part 51 have been satisfied; and

Revised License Condition g would state:

g. After weighing the environmental, economic, technical and other benefits of the facility against environmental costs and considering available alternatives, the issuance of this license (subject to the conditions for protection of the environment set forth herein) is in accordance with 10 CFR Part 51, of the Commission's regulations and all applicable requirements of said Part 51 have been satisfied; and

Current License Condition h states:

h. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and

(2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

Revised License Condition h would state:

h. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

Pursuant to 10 CFR 50.82(a)(2), as a result of the 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii) certifications submitted by the licensee, the 10 CFR Part 50 license for VY no longer authorize operation of the reactor. As such, reference to operation of the facility in License Conditions b, c, d, f, g, and h are inconsistent with the provisions of 10 CFR 50.82(a)(2). Therefore, the proposed changes provide consistency with 10 CFR 50.82(a)(2), and are acceptable.

3.4.3 Changes to License Condition 2.A

Current License Condition 2.A states:

A. Pursuant to Sections 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Entergy Nuclear Vermont Yankee, LLC to possess and use, and Entergy Nuclear Operations, Inc., to possess, use, and operate the facility as a utilization facility at the designated location on the Entergy Nuclear Vermont Yankee, LLC site.

Revised License Condition 2.A would state:

A. Pursuant to Sections 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Entergy Nuclear Vermont Yankee, LLC to possess and use, and Entergy Nuclear Operations, Inc., to possess and use the facility as a utilization facility at the designated location on the Entergy Nuclear Vermont Yankee, LLC site.

Pursuant to 10 CFR 50.82(a)(2), as a result of the 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii) certifications submitted by the licensee, the 10 CFR Part 50 license for VY no longer authorize operation of the reactors. As such, reference to operation of the facility in License Condition 2.A is inconsistent with the limitation imposed on the licensee by 10 CFR 50.82(a)(2). Therefore, the proposed change provides consistency with 10 CFR 50.82(a)(2), and is acceptable.

3.4.4 Changes to License Condition 2.B

Current License Condition 2.B states:

> B. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as supplemented and amended.

Revised License Condition 2.B would state:

> B. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as supplemented and amended.

The licensee states the proposed revision to this license condition is consistent with the restrictions of 10 CFR 50.82(a)(2) that no longer authorizes operation or emplacement of fuel in the reactor vessel at VY.

The proposed change removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel and eliminates the reference to use of the SNM for reactor operations and limits the possession of SNM to SNM "that was used" as reactor fuel at VY. Pursuant to 10 CFR 50.82(a)(2) the 10 CFR Part 50 license for VY no longer authorizes operation of the reactor. As such, the licensee has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as the licensee currently possesses the reactor fuel that was used for the past operations of the VY reactor. Based on the above, the proposed change to License Condition 2.B is consistent with the permanently shutdown status of VY and is, therefore, acceptable.

3.4.5 Changes to License Condition 2.C

Current License Condition 2.C states:

> C. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup,

sealed sources for calibration of reactor instrumentation and radiation monitoring equipment, and as fission detectors in amounts as required.

Revised License Condition 2.C would state:

C. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment, and as fission detectors in amounts as required.

The licensee proposed this license condition for revision to be consistent with the restriction of 10 CFR 50.82(a)(2) that VY is no longer authorized to operate. The proposed changes remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup but retains authorization to possess such sources previously used for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that VY is no longer authorized to operate and the continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. As such the proposed changes to the VY License Condition 2.C is consistent with the permanently shutdown status of the facility and is, therefore, acceptable.

3.4.6 Changes to License Condition 3.A

Current License Condition 3.A states:

A. <u>Maximum Power Level</u>

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.

Revised License Condition 3.A would state:

A. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and therefore power operation is no longer authorized.

The NRC staff reviewed the proposed deletion of License Condition 3.A and determined that power operation is no longer authorized at VY based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The licensee is not authorized to operate the VY at any power. Therefore, deletion of License Condition 3.A is appropriate and acceptable.

3.4.7 Changes to License Condition 3.F

Current License Condition 3.F states:

> F. Entergy Nuclear Operations, Inc. shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated January 13, 1978, and supplemental SERs, dated 9/12/79, 2/20/80, 4/15/80, 7/3/80, 10/24/80, 11/10/81, 1/13/83, 7/24/84, 3/25/86, 12/1/86, 12/8/89, 11/29/90, 8/30/95, 3/23/97, 6/9/97, 8/12/97, 3/6/98, 3/31/98, 9/2/98, and 2/24/99, subject to the following provisions:
>
> Entergy Nuclear Operations, Inc. may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Revised License Condition 3.F would state:

> F. This paragraph deleted by Amendment No. 263.

The licensee states that this license condition is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, "Fire protection," with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire and is no longer applicable at VY. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shutdown and defueled plant is not needed.

The NRC staff finds that License Condition 3.F for VY, is based on maintaining fire protection programs that provides reasonable assurance that the ability to achieve and maintain safe shutdown in the event of a fire in accordance with 10 CFR 50.48. Achieving and maintaining safe shutdown in the event of a fire is no longer applicable to the decommissioned fire protection programs at VY since the facility is permanently shutdown and the fuel has been removed from the reactor. However, elements of the fire protection program (e.g, License Condition 3.N, Mitigating Strategy) continue during decommissioning to address fire events that could result in radiological hazards. The regulation in 10 CFR 50.48(f) requires VY to address the potential for fires, which could result in a radiological hazard. The NRC staff concludes that the rule, which requires a fire protection program, is sufficient to ensure that a program is maintained. Therefore, a license condition that also requires fire protection programs for the permanently shutdown and defueled unit is redundant. Based on the above, the NRC staff concludes that reliance on 10 CFR 50.48(f) is appropriate and the licensee's request to eliminate License Condition 3.F is acceptable.

3.4.8 Changes to License Condition 3.K

Current License Condition 3.K states:

K. Minimum Critical Power Ratio

When operating at thermal power greater than 1593 megawatts thermal, the safety limit minimum critical power ratio (SLMCPR) shall be established by adding 0.02 to the cycle-specific SLMCPR value calculated using the NRC-approved methodologies documented in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," as amended, and documented in the Core Operating Limits Report.

Revised License Condition 3.K would state:

K. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and, therefore, power operation is no longer authorized.

The NRC staff concludes that power operation is no longer authorized at VY based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The proposed deletion is consistent with the lack of authority to operate at VY. Therefore, deletion of License Condition 3.K is appropriate and acceptable.

3.4.9 Changes to License Condition 3.L

Current license condition 3.L states:

L. Transient Testing

1. During the extended power uprate (EPU) power ascension test program and prior to exceeding 168 hours of plant operation at the nominal full EPU reactor power level, with feedwater and condensate flow rates stabilized at approximately the EPU full power level, Entergy Nuclear Operations, Inc. shall confirm through performance of transient testing that the loss of one condensate pump will not result in a complete loss of reactor feedwater.

2. Within 30 days at nominal full-power operation following successful performance of the test in (1) above, through performance of additional transient testing and/or analysis of the results of the testing conducted in (1) above, confirm that the loss of one reactor feedwater pump will not result in a reactor trip.

Revised License Condition 3.L would state:

L. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and, therefore, power operation is no longer authorized.

Based on its review, the NRC staff concludes that power operation is no longer authorized at VY based on the licensee's previous submission of 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The proposed deletion is consistent with the lack of authority to operate at VY. Therefore, deletion of License Condition 3.L is appropriate and acceptable.

3.4.10 Changes to License Condition 3.M

Current License Condition 3.M states:

M. Potential Adverse Flow Effects

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer).

1. The following requirements are placed on operation of the facility above the original licensed thermal power (OLTP) level of 1593 megawatts thermal (MWt):

   a. Entergy Nuclear Operations, Inc. shall monitor hourly the 32 main steam line (MSL) strain gages during power ascension above 1593 MWt for increasing pressure fluctuations in the steam lines.

   b. Entergy Nuclear Operations, Inc. shall hold the facility for 24 hours at 105%, 110%, and 115% of OLTP to collect data from the 32 MSL strain gages required by Condition M.1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall provide the evaluation to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission.

   c. If any frequency peak from the MSL strain gage data exceeds the limit curve established by Entergy Nuclear Operations, Inc. and submitted to the NRC staff prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall return the facility to a power level at which the limit curve is not exceeded. Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

d. In addition to evaluating the MSL strain gage data, Entergy Nuclear Operations, Inc. shall monitor reactor pressure vessel water level instrumentation or MSL piping accelerometers on an hourly basis during power ascension above OLTP. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data, Entergy Nuclear Operations, Inc. shall stop power ascension, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

e. Following start-up testing, Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis and provide that resolution to the NRC staff by facsimile or electronic transmission to the NRC project manager. If the uncertainties are not resolved within 90 days of issuance of the license amendment authorizing operation at 1912 MWt, Entergy Nuclear Operations, Inc. shall return the facility to OLTP.

2. As described in Entergy Nuclear Operations, Inc. letter BVY 05-084 dated September 14, 2005, Entergy Nuclear Operations, Inc. shall implement the following actions:

   a. Prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall install 32 additional strain gages on the main steam piping and shall enhance the data acquisition system in order to reduce the measurement uncertainty associated with the acoustic circuit model (ACM).

   b. In the event that acoustic signals are identified that challenge the limit curve during power ascension above OLTP, Entergy Nuclear Operations, Inc. shall evaluate dryer loads and re-establish the limit curve based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency.

   c. After reaching 120% of OLTP, Entergy Nuclear Operations, Inc. shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the steam dryer monitoring plan (SDMP) limit curve with the updated ACM load definition and revised instrument uncertainty, which will be provided to the NRC staff.

d. During power ascension above OLTP, if an engineering evaluation is required in accordance with the SDMP, Entergy Nuclear Operations, Inc. shall perform the structural analysis to address frequency uncertainties up to ±10% and assure that peak responses that fall within this uncertainty band are addressed.

e. Entergy Nuclear Operations, Inc. shall revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with General Electric Services Information Letter 644, Revision 1; and to identify the NRC Project Manager for the facility as the point of contact for providing SDMP information during power ascension.

f. Entergy Nuclear Operations, Inc. shall submit the final extended power uprate (EPU) steam dryer load definition for the facility to the NRC upon completion of the power ascension test program.

g. Entergy Nuclear Operations, Inc. shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC, including methodology for updating the limit curve, prior to initial power ascension above OLTP.

3. Entergy Nuclear Operations, Inc. shall prepare the EPU startup test procedure to include the (a) stress limit curve to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria are not satisfied; and (i) verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above OLTP. Entergy Nuclear Operations, Inc. shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above OLTP.

4. When operating above OLTP, the operating limits, required actions, and surveillances specified in the SDMP shall be met. The following key attributes of the SDMP shall not be made less restrictive without prior NRC approval:

a. During initial power ascension testing above OLTP, each test plateau increment shall be approximately 80 MWt;

b. Level 1 performance criteria; and

c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.

Changes to other aspects of the SDMP may be made in accordance with the guidance of NEI 99-04.

5. During each of the three scheduled refueling outages (beginning with the spring 2007 refueling outage), a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer, including flaws left "as is" and modifications.

6. The results of the visual inspections of the steam dryer conducted during the three scheduled refueling outages (beginning with the spring 2007 refueling outage) shall be reported to the NRC staff within 60 days following startup from the respective refueling outage. The results of the SDMP shall be submitted to the NRC staff in a report within 60 days following the completion of all EPU power ascension testing.

7. The requirements of paragraph 4 above for meeting the SDMP shall be implemented upon issuance of the EPU license amendment and shall continue until the completion of one full operating cycle at EPU. If an unacceptable structural flaw (due to fatigue) is detected during the subsequent visual inspection of the steam dryer, the requirements of paragraph 4 shall extend another full operating cycle until the visual inspection standard of no new flaws/flaw growth based on visual inspection is satisfied.

8. This license condition shall expire upon satisfaction of the requirements in paragraphs 5, 6, and 7 provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is due to fatigue.

Revised License Condition 3.M would state:

M. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted in its entirety because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and, therefore, power operation is no longer authorized.

The NRC staff reviewed the proposed deletion of License Condition 3.M and concludes that power operation is no longer authorized at VY based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The licensee is not authorized to operate at VY at any power. In addition, the steam dryer performs the function of removing moisture from the wet steam exiting the steam generators during reactor operation and serves no function in the permanently shutdown and defueled condition. Therefore, deletion of license condition 3.M is appropriate and acceptable.

3.4.11 Changes to License Condition 3.O

Current License Condition 3.O states:

> O. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

Revised License Condition 3.O would state:

> O. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and, therefore, power operation is no longer authorized.

The NRC staff reviewed the proposed deletion of License Condition 3.O and concludes that the DBA on which this order was based is not a postulated accident that can occur in the permanently defueled condition, rendering the actions required by the order and this license condition, unnecessary. In addition the NRC has approved ENO's request to rescind Order EA-06-137 on December 13, 2014 (ADAMS Accession No. ML14227A912). Therefore, deletion of License Condition 3.O, is appropriate and acceptable.

3.4.12 Changes to License Condition 3.R

Current License Condition 3.R states:

> R. Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. shall implement the most recent staff-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as the method to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix H. Any changes to the BWRVIP ISP capsule withdrawal schedule must be submitted for NRC staff review and approval. Any changes to the BWRVIP ISP capsule withdrawal schedule which affects the time of withdrawal of any surveillance capsules must be incorporated into the licensing basis. If any surveillance capsules are removed without the intent to test them, these capsules must be stored in a manner which maintains them in a condition which would support re-insertion into the reactor pressure vessel, if necessary.

Revised License Condition 3.R would state:

R. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and, therefore, power operation is no longer authorized. The requirements in Appendix H are only applicable to nuclear plants that are performing power operations in the reactor critical operating mode because: (a) this is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process, and (b) the requirements are set in place to provide assurance that the reactor pressure vessel (RPV) will maintain adequate levels of fracture toughness throughout the operating life of the reactor.

Based on its review of the proposed deletion, the NRC staff concludes that continued implementation of the applicable surveillance capsule testing and reporting requirements are no longer necessary for VY because: (a) power operation is no longer authorized at VY based on the 10 CFR 50.82(a)(2) certifications, and (b) from a fracture toughness perspective, the VY RPV will cease to be exposed to further irradiation by high energy neutrons or subjected to any high thermal stress environments, as induced by operating the RCS at an elevated temperature. The removal, testing, reporting, and storage requirements for reactor vessel surveillance capsules and their test specimens do not need to be implemented further since VY has permanently ceased power operations. Therefore, deletion of License Condition 3.R is appropriate and acceptable.

3.4.13 Changes to License Condition 3.S

Current License Condition 3.S states:

S. Steam Dryer License Condition

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. shall perform and implement the continuous parameter monitoring, moisture content monitoring, and visual inspections of the steam dryer described in BWRVIP-139-A, Final Report, dated July 2009, with the additional requirements included in the license condition specified herein.

1. The licensee shall inspect the steam dryer during refueling outage (RFO) 31. If the steam dryer inspection results from RFO 31 demonstrate that (a) any new cracking due to either fatigue or intergranular stress-corrosion cracking (IGSCC) is not significant (ie., is evaluated to be acceptable without repair), and (b) all previously identified indications (including the new indication identified in RFO 27) show no crack growth requiring repair, the licensee shall initiate a program to perform steam dryer inspections at least every third refueling outage.

2. If the steam dryer inspection during RFO 31 or any subsequent steam dryer inspection reveals new cracking due to either fatigue or IGSCC that

requires repair, or any growth of existing cracking that requires repair, the licensee shall inspect the steam dryer during each refueling outage.

3. The steam dryer inspection during RFO 31 and all future steam dryer inspections shall meet the guidelines in BWRVIP-139-A with the exception of the inspection interval specified in BWRVI P-139-A. The results of the steam dryer inspection beginning with RFO 31 and continuing for all future inspections shall be submitted to the NRC within 6 months of each steam dryer inspection completion.

4. This license condition shall continue for the full term of the period of extended operation unless duly amended.

Revised License Condition 3.S would state:

S. This paragraph deleted by Amendment No. 263.

The licensee stated that this license condition can be deleted because VY is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and, therefore, power operation is no longer authorized.

The NRC staff reviewed the proposed deletion of License Condition 3.S. The steam dryer performs the function of removing moisture from the wet steam exiting the steam separators during reactor operation and serves no function in the shutdown condition. As noted above, power operation is no longer authorized at VY based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. Therefore, deletion of License Condition 3.S for the steam dryer monitoring and inspection program is appropriate and acceptable.

3.4.13 Changes to License Condition 4

Current License Condition 4 states:

4. This renewed operating license is effective as of the date of issuance and shall expire at midnight on March 21, 2032.

Revised License Condition 4 would state:

4. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

The licensee stated that this license condition is revised to reflect the permanently defueled condition at the facility in accordance with 10 CFR 50.82(a)(2). ENO has proposed that this license condition be revised to conform to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of VY until the Commission notifies the licensee in writing that the license is terminated.

The NRC staff reviewed the proposed Licensee Condition 4. The current License Condition 4, which documents the date of the expiration of the renewed license, is no longer necessary for a permanently shutdown condition of the plant in the process of decommissioning. The revised License Condition 4 documents the current condition of the plant and summarizes the actions and requirements applicable to the facility by 10 CFR 50.51. The revised License Condition 4 is consistent with the regulatory requirements applicable to the facility in the permanently shutdown and defueled condition. Therefore, the revised License Condition 4 is appropriate and acceptable.

3.5 Continued Safe Operations of the SFP

As stated above, 10 CFR 50.51(b) requires licensees to take actions necessary to decommission and decontaminate the facility and continue to maintain the facility in a safe condition. The structures and components required to maintain the facility during the decommissioning period may remain operational beyond the normal licensed operating period of 40 years and the 20 year renewal period previously authorized at VY. Under the provisions of 10 CFR 50.82, the licensee must complete decommissioning within 60 years of permanent cessation of operations. Long lived passive structures and components, such as those used to maintain the SFP, may remain in operation up to 60 years after the permanent cessation of operations.

The licensee currently has a neutron absorber monitoring program for the SFP in its UFSAR as a part of License Renewal Commitment 52. This program requires the license to perform testing either via coupon measurement and/or via neutron attenuation testing using an in situ method every 10 years. The testing includes periodic verification of boron loss through areal density measurement of coupons or through direct in situ techniques, such as measurement of boron areal density, measurement of geometric changes in the material (blistering, pitting, and bulging), and detection of gaps through blackness testing. The NRC reviewed and approved this program and the UFSAR description of the program as part of the license renewal review. As described in NRC Regulatory Issue Summary 2014-09: "Maintaining the Effectiveness of License Renewal Aging Management Programs," dated August 6, 2014 (ADAMS Accession No. ML14058A398), holders of licenses issued under 10 CFR Part 54 are required to maintain the effectiveness of their aging management programs and activities. The licensee is obligated to maintain the effectiveness of the neutron absorber monitoring program described in its UFSAR. As such, the licensee's existing neutron absorbing material monitoring program, described in the UFSAR, is adequate for managing the effects of aging in the SFP.

The first test using this program was performed in September and October of 2014 using the Boron-10 Areal Density Gage for Evaluating Racks (BADGER) method. The measured panels had a measured average panel areal density values above the minimum required areal density, which is an acceptable result.

In addition, in its supplemental letter dated May 4, 2015, the licensee stated that all fuel in the SFP is scheduled to be removed from the SFP to dry fuel storage by December 31, 2020, which is 4 years before the next neutron attenuation test is due.

The changes proposed by the licensee for the Section 3/4.12, "Spent Fuel Storage" TSs do not affect the design or use of the existing fuel racks, and therefore no criticality analysis was made.

The systems needed to keep the fuel in the SFP in a subcritical condition are also kept intact. TS 3/4.12.C, "Fuel Storage Pool Water Level" and TS 3/4.12.H, "Spent Fuel Pool Water Temperature" requirements for safe storage of spent fuel are retained. The NRC staff concludes that either removing all fuel by December 31, 2020, or thereafter continuing the current program as stated in the UFSAR, is acceptable to ensure continued safe operation of the SFP during the decommissioning period.

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 the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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, which was published in the *Federal Register* on February 17, 2015 (80 FR 8358). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10)(ii). 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) there is reasonable assurance that 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: K. Bucholtz
M. Greenleaf
E. Dickson
S. Shaikh
D. Cunanan

Date: October 7, 2015

October 7, 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 FOR DEFUELED TECHNICAL SPECIFICATIONS AND REVISED LICENSE CONDITIONS FOR PERMANENTLY DEFUELED CONDITION (CAC NO. MF3714)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 263 to Renewed Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station (VY), in response to your application dated March 28, 2014, as supplemented by letters dated April 24, June 9, June 11, and August 13, 2014; and May 4, 2015.

The amendment revises the renewed facility operating license and the associated technical specifications to conform to the permanent shutdown and defueled status at VY.

A copy of the related 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. 263 to DPR-28
2. Safety Evaluation

cc w/encls: Distribution via Listserv

<u>DISTRIBUTION</u>: See next page

ADAMS Accession No.: ML15117A551 *** via memorandum** ****via email**

OFFICE	NRR/DORL/LPLIV-2/PM	NRR/DORL/LPLIV-2/LA	NRR/DSS/STSB/BC*	NRR/DSS/SBPB/BC*
NAME	JKim	PBlechman	RElliott	GCasto
DATE	9/17/2015	9/16/2015	11/25/2014	12/04/2014
OFFICE	NRR/DSS/SCVB/BC*	NRR/DRA/ARCB/BC*	NRR/DSS/SRXB/BC*	OGC – NLO w/comments
NAME	RDennig	(*JDozier for*) UShoop	CJackson	MYoung
DATE	1/15/2015	2/27/2015	8/21/2015	10/01/2015
OFFICE	NMSS/DUWP/RDB/BC**	NRR/DORL/LPLIV-2/BC	NRR/DORL/LPLIV-2/PM	
NAME	BWatson	MKhanna	JKim	
DATE	10/06/2015	10/05/2015	10/07/2015	

OFFICIAL RECORD COPY

Letter to Vice President, Operations from James Kim dated: October 7, 2015

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION – ISSUANCE OF AMENDMENT FOR PERMANENTLY SHUTDOWN AND DEFUELED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS (CAC NO. MF3714)

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