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BVY 13-096

October 31, 2013

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

SUBJECT: Technical Specifications Proposed Change No. 307
Revision to Mitigation Strategy License Condition and Technical Specification Administrative Controls for Permanently Defueled Condition
Vermont Yankee Nuclear Power Station
Docket No. 50-271
License No. DPR-28

REFERENCES:

1. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Notification of Permanent Cessation of Power Operations," BVY 13-079, dated September 23, 2013

2. Letter, Entergy Nuclear Operations, Inc. to USNRC "Request for Approval of Certified Fuel Handler Training Program," BVY 13-095, dated October 31, 2013

Dear Sir or Madam:

In accordance with 10CFR50.90, Entergy Nuclear Operations, Inc. (ENO) is proposing an amendment to Renewed Facility Operating License (OL) DPR-28 for Vermont Yankee Nuclear Power Station (VY).

In Reference 1, ENO notified the NRC that it has decided to permanently cease operations of VY at the end of the current operating cycle. Once certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii), per 10 CFR 50.82(a)(2), the 10 CFR Part 50 license no longer will permit operation of the reactor or placement of fuel in the reactor vessel. The basis for the proposed amendment is that certain license conditions and administrative controls may be revised or removed to reflect the permanently defueled condition.

Part of this amendment request is to eliminate the Mitigation Strategy License Condition from the VY OL. This request also proposes changes to the staffing and training requirements for the VY staff contained in Section 6.0, Administrative Controls, of the VY Technical Specifications (TS). Reference 2 was submitted proposing a Certified Fuel Handler training program for NRC approval. Additional changes are proposed to certain required reports and programs contained in Section 6.0 that will no longer be applicable once VY is permanently defueled.

A001
NRR

ENO has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the State of Vermont, Department of Public Service.

Attachment 1 to this letter provides a detailed description and evaluation of the proposed change. Attachment 2 contains a markup of the current OL and TS pages. Attachment 3 contains the retyped OL and TS pages.

ENO requests review and approval of this proposed license amendment by November 1, 2014 and a 60 day implementation period from the effective date of the amendment. ENO requests that the approved amendment become effective following NRC approval of the Certified Fuel Handler training program (Reference 2) and submittal of the certifications required by 10 CFR 50.82(a)(1).

There are no new regulatory commitments made in this letter.

If you have any questions on this transmittal, please contact Mr. Coley Chappell at 802-451-3374.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 31, 2013.

Sincerely,

CJW/plc

Attachments:

1. Description and Evaluation of the Proposed Changes
2. Markup of the Current Operating License and Technical Specification Pages
3. Retyped Operating License and Technical Specification Pages

cc: Mr. William M. Dean
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Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Change 307

Description and Evaluation of Proposed Changes

1. SUMMARY DESCRIPTION

On August 27, 2013, Entergy Nuclear Operations, Inc. (ENO) announced that Vermont Yankee Nuclear Power Station (VY) would be permanently retired at the end of the current operating cycle. In Reference 1, ENO provided formal notification of the intention to permanently cease power operations of VY in accordance with 10 CFR 50.82(a)(1)(i).

This evaluation supports a request to amend Renewed Facility Operating License (OL) DPR-28 for VY. The proposed changes would revise and remove certain requirements contained within Section 6.0, Administrative Controls, of the VY Technical Specifications (TS) and remove the Mitigation Strategy License Condition from the OL. The TS and OL requirements being changed would not be applicable once it has been certified that all fuel has permanently been removed from the VY reactor in accordance with 10 CFR 50.82(a)(1)(ii). Once the certifications for permanent cessation of operations and permanent fuel removal from the reactor vessel are made, the 10 CFR Part 50 license for VY no longer will authorize operation of the reactor or placement of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2).

The changes proposed by this amendment would not be effective until the certification of permanent removal of fuel from the reactor vessel has been submitted to the NRC and the NRC has approved the VY Certified Fuel Handler training program submitted in Reference 2.

2. DETAILED DESCRIPTION AND BASIS FOR THE CHANGES

The following table identifies each section that is being changed, the proposed changes, and the basis for the changes:

<table>
<tr><th colspan="2">Proposed Change to VY License Condition 3.N, Mitigation Strategy License Condition</th></tr>
<tr><td><u>Current License Condition 3.N</u>

N. Mitigation Strategy License Condition

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</td><td><u>Proposed License Condition 3.N</u>

N. Deleted</td></tr>
</table>

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	

Basis

This section is proposed for deletion in its entirety. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, the mitigation strategy license condition will no longer be required.

The NRC issued this license condition on August 23, 2007, to incorporate the requirements for the Interim Compensatory Measures (ICM) Order EA-02-026, Section B.5.b mitigation strategies (dated February 25, 2002). Subsequently, 10 CFR 50.54(hh)(2) became effective on May 26, 2009. This section provides mitigation strategies and response procedure requirements for loss of large areas of the plant due to explosions or fire. However, as stated in 10 CFR 50.54(hh)(3), this section does not apply to a defueled reactor that has submitted the certification for permanent removal of fuel under 10 CFR 50.82(a).

On November 28, 2011, the NRC issued a letter to rescind Item B.5.b of the ICM Order EA-02-026. Therefore, the ICM Order does not apply to VY and 10 CFR 50.54(hh) will not apply to VY once the certifications required by 10 CFR 50.82(a)(1) have been submitted.

Proposed Changes to VY Technical Specification Section 6.0, Administrative Controls

6.1 Responsibility

Current TS 6.1.C	Proposed TS 6.1.C
The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in plant startup or normal operation, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in cold shutdown or refueling with fuel in the reactor, an individual with an active SRO license or	The shift supervisor shall be responsible for the shift command function.

Reactor Operator license shall be designated to assume the control room command function.	

Basis

This section identifies the responsibilities for the control room command function associated with Modes of plant operation, and is based on personnel positions and qualifications for an operating plant. It identifies the need for a delegation of authority for command in an operating plant when the principal assignee leaves the control room.

This section is being changed to eliminate the Mode dependency for this function and personnel qualifications associated with an operating plant. The proposed change establishes the shift supervisor as having command of the shift. Delegation of command is unnecessary once VY is in the permanently defueled condition with fuel in the spent fuel pool. Any event involving loss of pool cooling would evolve slowly enough that no immediate response would be required to protect the health and safety of the public or station personnel.

6.2 Organization

<u>Current TS 6.2.A, Onsite and Offsite Organizations</u>	<u>Proposed TS 6.2.A, Onsite and Offsite Organizations</u>
Organizations shall be established for unit operation and corporate management. These organizations shall include the positions for activities affecting safety of the nuclear power plant.	Organizations shall be established for unit operation and corporate management. These organizations shall include the positions for activities affecting safety of the nuclear fuel.
2. The plant manager shall be responsible for overall unit safe operation and shall have control over those on-site activities necessary for safe operation and maintenance of the plant.	2. The plant manager shall be responsible for overall unit safe operation and shall have control over those on-site activities necessary for safe storage and maintenance of the nuclear fuel.
3. The site vice president 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 nuclear safety.	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 operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate on-site	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

manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.	individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

Basis

The introduction to this section identifies that organizational positions are established that are responsible for the safety of the nuclear plant. This is changed to require that positions be established that are responsible for the safe handling and storage of nuclear fuel. This change removes the implication that VY can return to operation once the certifications required by 10 CFR 50.82(a)(1) are submitted to the NRC.

Although the term "unit" remains unchanged throughout these proposed changes, whereas "unit" was used in reference to an operating reactor, now "unit" will refer to the systems, structures and components (SSC) required to support spent fuel storage and fuel handling operations. "Unit" will also refer to the VY facility in regards to activities occurring on the site, including those in support of decommissioning. Similarly, the term "plant" remains unchanged, although it will now also refer to the SSCs required to support spent fuel storage and fuel handling operations.

The terms "safe storage and maintenance of nuclear fuel" and "safe management of nuclear fuel" are considered analogous to "nuclear safety" for a plant that will be in the permanently defueled condition. Proposed changes to replace "nuclear safety" with one of these analogues serves to narrow the focus of nuclear safety concerns to the nuclear fuel.

TS 6.2.A.1 - No changes are proposed to this specification.

TS 6.2.A.2 – This section identifies the organizational position responsible for the safe operation of the plant, and for control of activities necessary for the safe operation and maintenance of the plant.

To reflect the change in safety concerns from an operating plant to a permanently defueled plant, the responsibility for control of activities necessary for the safe operation and maintenance of the plant is changed to the responsibility for safe storage and maintenance of the nuclear fuel.

TS 6.2.A.3 - This section identifies the organizational position responsible for overall nuclear plant safety.

To reflect the change in safety concerns from an operating plant to a permanently defueled plant, the responsibility for ensuring nuclear safety is changed to the responsibility for ensuring safe management of nuclear fuel. The assignment of this responsibility is changed from the VY site vice president to a specified corporate officer. This change provides ENO the flexibility to assign overall responsibility to a corporate officer position other than a site vice president. The site vice president is considered a corporate officer position. This position has no qualification requirements beyond the applicable requirements established in ANSI/ANS 3.1-1978. The revised specification is consistent with TS 5.2.1.c of NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," Revision 4 (Reference 3).

TS 6.2.A.4 - This paragraph addresses the requirement for organizational independence of the personnel who train the operations staff, health physics personnel and quality assurance personnel from operating pressures.

This is changed to replace "operating staff' with "Certified Fuel Handlers" and to replace "their independence from operating pressures" to "their ability to perform their assigned functions." These changes reflect the changed function of the previous operating staff to a focus on safe handling and storage of nuclear fuel, and to remove the implication that VY can return to operation once the certifications required by 10 CFR 50.82(a)(1) are submitted to the NRC.

Current TS 6.2.B, Unit Staff	Proposed TS 6.2.B, Unit Staff
The unit staff organization shall include the following:	The unit staff organization shall include the following:
1. A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned during Plant Startup and Normal Operation.	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 licensed Reactor Operator (RO) or one licensed Senior Reactor Operator (SRO) shall be present in the control room when fuel is in the reactor.	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. When the unit is in Plant Startup or Normal Operation, at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, shall be present in the control room.	3. All fuel handling operations shall be directly supervised by a Certified Fuel Handler.
4. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.B.1 and 6.2.B.8 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.	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 when there is fuel in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.	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. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).	6. Deleted
7. The operations manager or an assistant operations manager shall hold an SRO license.	7. The shift supervisor shall be a Certified Fuel Handler.
8. While the unit is in Plant Startup or Normal Operation an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operations of the unit. This individual shall meet the qualifications specified by ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Rev. 3, 2000.	8. Deleted

Basis

TS 6.2.B.1 - This paragraph addresses that one auxiliary nuclear operator must be assigned to the operating shift whenever fuel is in the reactor.

Since this can never occur again at VY once the certifications required by 10 CFR 50.82(a)(1) are submitted to the NRC, the minimum requirement is changed to a minimum crew compliment of one shift supervisor and one Non-certified Operator. This reflects the reduced number of systems, compared to an operating reactor, required to provide and support spent fuel pool cooling and monitor spent fuel pool parameters, such as pool level and temperature, while still maintaining the ability to ensure spent fuel handling operations are carried out in a safe manner. Moreover, the spectrum of credible accidents and operational events, and the quantity and complexity of activities required for safety has been greatly reduced from that at an operating plant. The shift supervisor will be qualified as a Certified Fuel Handler in accordance with new paragraph 6.2.B.7. In this position, this individual will retain command and control responsibility for operational decisions and will be responsible for the functions required for event reporting and emergency response.

TS 6.2.B.2 - This paragraph establishes the requirement for one licensed Reactor Operator to be in the control room when fuel is in the reactor.

VY will not be required to have operators licensed pursuant to 10 CFR 55 once the certifications of 10 CFR 50.82(a)(1) have been submitted to the NRC. As a result, TS 6.2.B.2 will not apply.

This paragraph is changed to reflect the requirement for having one qualified watch stander (either a Non-certified Operator or Certified Fuel Handler) in the control room when fuel is stored in the spent fuel pool. This reflects the reduced requirement for control room personnel training and qualification for a plant authorized for nuclear fuel storage only. VY has submitted a Certified Fuel Handler training program for NRC approval in Reference 2. The training and qualification for the Non-certified Operator will be determined in accordance with the systems approach to training (SAT) as defined in 10 CFR 55.4. This process ensures that the Non-certified Operator will be qualified to perform the functions necessary to monitor and ensure safe storage of fuel. The SAT process requires (1) systematic analysis of the jobs to be performed, (2) learning objectives derived from the analysis which describe desired performance after training, (3) training design and implementation based on the learning objectives, (4) evaluation of trainee mastery of the objectives during training, and (5) evaluation and revision of the training based on the performance of trained personnel in the job setting. There will be a sufficient number of individuals qualified as Certified Fuel Handlers to staff the plant twenty four hours a day, seven days a week. Additional on-shift staffing will be provided to satisfy applicable security, fire protection, and emergency preparedness requirements.

The control room will remain the physical center of the command function. However, since control of activities may be performed either remotely from the control room or locally in the plant, the location of the command center is functionally where the shift supervisor is located, in accordance with proposed TS 6.1.C. Activities that could be performed from the control room that have the potential to affect forced cooling of spent nuclear fuel include starting and stopping cooling water pumps, as well as changing the electrical power distribution system alignment.

All spent fuel handling activities are performed locally at the spent fuel pool. Indications and/or alarms are also received in the control room that would be indicative of spent fuel pool abnormalities. The shift supervisor is responsible for directing response to those abnormalities, from either the control room or local to the spent fuel pool, in accordance with applicable response procedures.

For any conditions, incidents, or events that occur when the Non-certified Operator is in the control room alone and are not within the scope of qualifications that are possessed by the Non-certified Operator, the shift supervisor will be immediately contacted for direction by phone, radio, and/or plant page system. This philosophy is deemed acceptable because the necessity to render immediate actions to protect the health and safety of the public is not challenged.

TS 6.2.B.3 - This paragraph establishes the requirement for at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, to be present in the control room during Plant Startup or Normal Operation.

VY will not be required to have operators licensed pursuant to 10 CFR 55 once the certifications of 10 CFR 50.82(a)(1) have been submitted to the NRC. As a result, TS 6.2.B.3 will not apply.

This paragraph is changed to establish the requirement for having oversight of fuel handling operations performed by a Certified Fuel Handler. Fuel moves and heavy load moves that could affect the safe handling and storage of nuclear fuel would be approved by the shift supervisor. Proposed TS 6.2.B.7 requires the shift supervisor to be a Certified Fuel Handler.

TS 6.2.B.4 - This paragraph addresses the conditions under which the minimum shift compliment may be reduced. It contains a reference to 10 CFR 50.54(m) which establishes the minimum requirements for a licensed operating staff for facility operation. It also allows for shift crew composition to be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.B.1 and 6.2.B.8 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

The references to 10 CFR 50.54(m) are removed since VY will not return to operation in the future once the certifications required by 10 CFR 50.82(a)(1) are submitted to the NRC, and the requirement for licensed operating personnel will no longer be required to protect public health and safety. ENO has submitted a separate request for exemption from 10 CFR 50.54(m) for VY in Reference 4. Reference to TS 6.2.B.8 is removed to be consistent with the proposed change to delete the Specification. Additional provisions are added to ensure that shift crew composition is not below the minimum requirements when fuel movements are in progress, movements of loads over fuel are in progress or shift turnover is in progress.

TS 6.2.B.5 - This paragraph establishes the requirement for a person qualified in radiation protection procedures to be onsite when fuel is in the reactor. This paragraph also allows for the position to be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

This requirement is being deleted because once the fuel is certified removed from the reactor in accordance with 10 CFR 50.82(a)(1)(ii), VY will be prohibited by 10 CFR 50.82(a)(2) from placing fuel back into the reactor vessel. Therefore, this requirement will no longer be applicable.

This requirement is being replaced with a requirement for an individual qualified in radiation protection procedures to be present on-site during the movement of fuel and during the movement of loads over fuel.

The radiation protection manager is not required to be on shift; although the radiation protection manager is part of the normal facility staff. Per proposed TS 6.2.B.1 and TS 6.2.B.5, there is no requirement for a qualified radiation protection manager to be a part of the minimum shift crew. Radiation protection (RP) technical oversight during fuel handling activities is provided by facility or supplemental RP personnel as specified in applicable RP and fuel handling procedures.

TS 6.2.B.6 - This paragraph establishes requirements to have procedures in place to limit the working hours of staff members who perform safety related functions.

This paragraph is deleted because the requirements contained within have been incorporated into 10 CFR 26, Subpart I – Managing Fatigue. The requirements of 10 CFR 26, Subpart I will continue to be applicable to VY staff, as defined by 10 CFR 26.201 and 10 CFR 26.4, in the permanently defueled condition. Therefore, the requirements of TS 6.2.B.6 are redundant and can be deleted.

TS 6.2.B.7 - This paragraph establishes the requirement for the operations manager, or an assistant operations manager, to hold a Senior Reactor Operator (SRO) license.

This paragraph is being revised to replace the requirement with a requirement that the shift supervisor shall be a Certified Fuel Handler. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the requirements of 10 CFR 50.54(m) will no longer be applicable because the VY Part 50 license no longer will authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel. These certifications also obviate the need for the operators' licenses specified in 10 CFR 55. Therefore, there is no longer a need for operations management staff to hold a SRO license. Replacing this with a requirement that the shift supervisor shall be a Certified Fuel Handler ensures that the senior individual on shift is appropriately trained and qualified, in accordance with the NRC-approved Certified Fuel Handler training program, to supervise shift activities. ENO has submitted a separate request for exemption from 10 CFR 50.54(m) for VY in Reference 4.

The VY management structure will not require positions above the shift supervisor to be a Certified Fuel Handler or attend equivalent training. VY has determined that once the plant is permanently shutdown and defueled, the time available to mitigate credible events is expected to be greater than that for current design basis events. As such, management oversight of the plant can be performed by individuals meeting the applicable requirements of ANSI/ANS 3.1-1978 (as required by TS 6.2.C.1) and need not be qualified as Certified Fuel Handlers.

<u>TS 6.2.B.8</u> - This paragraph establishes the requirements for the Shift Technical Advisor (STA) position.

This paragraph is deleted to remove the requirements for the STA since that position is only required for a plant authorized for power operations. Once the certifications required by 10 CFR 50.82(a)(1) have been submitted, the requirements of this specification will no longer be applicable because the VY Part 50 license no longer will authorize operation of the reactor or emplacement or retention of fuel in the reactor vessel.

<u>Current TS 6.2.C, Unit Staff Qualifications</u>	<u>Proposed TS 6.2.C, Unit Staff Qualifications</u>
2. For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 6.2.C.1, perform the functions described in 10 CFR 50.54(m).	2. An NRC approved training and retraining program for Certified Fuel Handlers shall be maintained.

Basis

<u>TS 6.2.C.1</u> - No changes are proposed to this paragraph.

<u>TS 6.2.C.2</u> - This paragraph defines SROs and ROs as the individuals who perform the functions defined in 10 CFR 50.54(m).

This paragraph is being deleted because neither 10 CFR 50.54(m) nor the requirement for licensed operators per 10 CFR 54 apply following submittal of the certifications required by 10 CFR 50.82(a)(1). ENO has submitted a separate request for exemption from 10 CFR 50.54(m) for VY in Reference 4.

A new TS 6.2.C.2 is being added to require that an NRC approved training and retraining program for the Certified Fuel Handlers shall be maintained. The Certified Fuel Handler training program ensures that the qualifications of fuel handlers are commensurate with the tasks to be performed and the conditions requiring response. 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," requires training programs to be derived using a systems approach to training (SAT) as defined in 10 CFR 55.4. Although the requirements of 10 CFR 50.120 apply to holders of an operating license issued under Part 50, and the VY license will no longer authorize operation following submittal of the certifications required by 10 CFR 50.82(a)(1), the Certified Fuel Handler training program nonetheless aligns with those requirements. The Certified Fuel Handler training program provides adequate confidence that appropriate SAT based training of personnel who will perform the duties of a Certified Fuel Handler is conducted to ensure the facility is maintained in a safe and stable condition.

6.3 Action To Be Taken If A Safety Limit Is Exceeded

Current TS 6.3	Proposed TS 6.3
Applies to administrative action to be followed in the event a safety limit is exceeded. If a safety limit is exceeded, the reactor shall be shutdown immediately.	Deleted

Basis

This paragraph defines the requirement to immediately shutdown the reactor if a safety limit is exceeded.

This paragraph is proposed to be deleted. Once VY submits the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2). Therefore, this Specification will not be needed once VY is in the permanently defueled condition.

6.4 Procedures

Current TS 6.4	Proposed TS 6.4
Written procedures shall be established, implemented, and maintained covering the following activities:	Written procedures shall be established, implemented, and maintained covering the following activities:
B. Refueling operations.	B. Fuel handling operations.
C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected Primary System leaks and abnormal reactivity changes.	C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components.
E. Preventive and corrective maintenance operations which could have an effect on	E. Preventive and corrective maintenance operations which could have an effect on

the safety of the reactor.	the safety of the nuclear fuel.

Basis

This paragraph provides a description and requirements regarding administration of written procedures. TS 6.4 will remain applicable with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition. Relevant procedures drawings and instructions will continue to be controlled per 10 CFR 50, Appendix B, Criterion VI, "Document Control." Activities involving security and emergency planning and preparedness will continue to be controlled by procedure.

<u>TS 6.4.B</u> - This TS is being revised to specify that procedures are required for fuel handling operations, rather than refueling, because refueling of the reactor will be prohibited by the 10 CFR Part 50 license once the certifications required by 10 CFR 50.82(a)(1) have been submitted to the NRC. Procedures governing fuel handling operations will provide the guidance necessary to ensure safe handling of spent fuel in the spent fuel pool and transfer from the spent fuel pool to dry fuel storage casks. Procedures governing responses to fuel handling accidents, personnel injuries, spent fuel pool events and external events provide the necessary guidance to mitigate the consequences of such events. No change to VY's response to a fuel handling accident is being proposed in this submittal.

<u>TS 6.4.C</u> - This TS is being revised to reflect that once the reactor is permanently defueled, reactivity changes in the reactor vessel will no longer be possible and the Primary Systems that provide reactor core cooling will no longer be required. Therefore, a requirement to have procedural direction to address reactivity changes or suspected Primary System leakage does not apply in the permanently defueled condition.

<u>TS 6.4.E</u> - There is no need to perform maintenance on the reactor in the permanently defueled condition to ensure safety because operation of the reactor will be prohibited by the Part 50 license once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Therefore, the requirement to have procedures in place to perform maintenance is not applicable in the permanently defueled condition. This Specification is being revised to require that procedures for performing maintenance that could affect the safety of the nuclear fuel be in place. This revised requirement will ensure the safety of the nuclear fuel during spent fuel pool storage and during fuel handling operations.

There are no changes proposed to TS 6.4.A, D and F-I.

6.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4. <u>Current TS 6.6.C</u> <u>Core Operating Limits Report</u> The core operating limits shall be established	The following reports shall be submitted in accordance with 10 CFR 50.4. <u>Proposed TS 6.6.C</u> Deleted

and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following: 1. The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a, 2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a, 3. The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a and 3.11.B, and 4. The Power/Flow Exclusion Region for Specifications 3.6.J.1.a and 3.6.J.1.b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in: *[list of references not copied here - see TS markup in Attachment 2]* The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC.	
Basis	
TS 6.6.C, Core Operating Limits Report, is being deleted because the Core Operating Limits Report pertains only to an activity that does not apply in a permanently defueled condition and after the certifications required by 10 CFR 50.82(a)(1) have been submitted to the NRC.	
6.7 Programs and Manuals	
The following programs shall be established, implemented and maintained:	The following programs shall be established, implemented and maintained:

Current TS 6.7.A	Proposed TS 6.7.A
A. INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels will be implemented. This program shall include the following: 1. Provisions establishing preventive maintenance and periodic visual inspection requirements. 2. System leakage inspections, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are: (1) Residual Heat Removal, (2) Core Spray, (3) Reactor Water Cleanup, (4) HPCI, (5) RCIC, and (6) Sampling Systems.	A. Deleted
Current TS 6.7.B.1.b, Offsite Dose Calculation Manual 1. Licensee initiated changes to the ODCM: b. Shall become effective upon review by PORC and approved by the plant manager.	Proposed TS 6.7.B.1.b, Offsite Dose Calculation Manual 1. Licensee initiated changes to the ODCM: b. Shall become effective upon approval by the plant manager.
Current TS 6.7.C, Primary Containment Leakage Rate Testing Program A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program," dated September 1995, as modified by the following: • The first Type A test after the April 1995	Proposed TS 6.7.C Deleted

Type A test shall be performed prior to startup from the April 2010 refuel outage. (This is an exception to Section 9.2.3 of NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J.") • The leakage contributions from the main steam pathways are excluded from the sum of the leakage rates from Type B and C tests specified in (1) Section III.B of 10CFR50, Appendix J – Option B; (2) Section 6.4.4 of ANSI/ANS 56.8-1994; and (3) Section 10.2 of NEI 94-01, Rev. 0. • The leakage contributions from the main steam pathways are excluded from the overall integrated leakage rate from Type A tests specified in (1) Section III.A of 10CFR50, Appendix J – Option B; (2) Section 3.2 of ANSI/ANS 56.8-1994; and (3) Sections 8.0 and 9.0 of NEI 94-01, Rev. 0. The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 44 psig. The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.8% of primary containment air weight per day. Leakage rate acceptance criteria are: 1. Primary containment leakage rate acceptance criterion ≤ 1.0 La. 2. The as-left primary containment integrated leakage rate test (Type A test) acceptance criterion is ≤ 0.75 La. 3. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from the main steam pathways) is ≤ 0.6 La, calculated on a maximum pathway basis, prior to entering a mode of operation where primary containment integrity is required. 4. The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage	

contributions from the main steam pathways) is ≤ 0.6 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required. 5. Airlock overall leakage rate acceptance criterion is ≤ 0.10 La when tested at ≥ Pa. The provision of SR for 4.0.2 for Surveillance Frequency does not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.	

Basis

TS 6.7, Program and Manuals, provides a description and requirements regarding programs and manuals that are to be established, implemented, and maintained. TS 6.7 will remain applicable with the reactor permanently defueled. As such, it is being retained and revised to reflect a permanently defueled condition.

TS 6.7.A, Integrity of Systems Outside Containment - This was established to minimize leakage from portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The program is being eliminated since these conditions can no longer exist for a permanently defueled plant.

TS 6.7.B.1.b, Offsite Dose Calculation Manual - This change eliminates the requirement for the Plant Operations Review Committee (PORC) to review licensee initiated changes to the Offsite Dose Calculation Manual (ODCM). The plant manager will continue to approve licensee initiated changes to the ODCM. This change makes this Specification consistent with the equivalent specification (TS 5.5.1) in NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," Revision 4. Additionally, reviews of ODCM changes are required to be reviewed by the On-Site Safety Review Committee (OSRC) in accordance with Entergy procedure EN-OM-119, "On-Site Safety Review Committee." The PORC and OSRC are equivalent.

TS 6.7.C, Primary Containment Leakage Rate Testing Program - This program is being deleted because the Primary Containment Leakage Rate Testing Program pertains only to reactor support systems that do not apply in a permanently defueled condition. Therefore, the need for leakage rate testing of primary containment is no longer applicable.

3. REGULATORY EVALUATION

3.1 APPLICABLE REGULATORY REQUIREMENT/CRITERIA

10 CFR 50.82(a)(1) requires that when a licensee has determined to permanently cease operations the licensee shall, within 30 days, submit a written certification to the NRC, consistent with the requirements of 10 CFR 50.4(b)(8), and once fuel has been permanently removed from the reactor vessel, the licensee shall submit a written certification to the NRC that meets the requirements of 10 CFR 50.4(b)(9). On August 27, 2013, Entergy Nuclear Operations, Inc. (ENO) announced that VY would be retired in the fourth quarter of 2014, the exact date to be determined. In Reference 1, ENO provided formal notification of the intention to permanently cease power operations of VY. VY recognizes that approval of these proposed changes is contingent upon the submittal of the certifications required by 10 CFR 50.82(a)(1).

10 CFR 50.82(a)(2) states "Upon docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, or when a final legally effective order to permanently cease operations has come into effect, the 10 CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel."

10 CFR 50.36 establishes the requirements for Technical Specifications. 50.36(c)(5), Administrative Controls, identifies that an Administrative Controls section shall be included in the Technical Specifications and shall include provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. This amendment request is proposing changes to the Administrative Controls section consistent with the pending decommissioning status of the plant. This request applies the principles identified in 50.36(c)(6), Decommissioning, for a facility which has submitted certifications required by 50.82(a)(1) and proposes changes to the Administrative Controls appropriate for the VY permanently defueled condition. As 10 CFR 50.36(c)(6) states, this type of change should be considered on a case-by-case basis.

10 CFR 50.54(m) establishes the requirements for having Reactor Operators and Senior Reactor Operators licensed in accordance with Part 55 based on plant conditions. Based on the impending permanent cessation of operation for VY, the requirements of this section will no longer apply once the certifications required by 10 CFR 50.82(a)(1) have been submitted to the NRC and it will be permissible to remove those positions from the Technical Specifications. ENO has submitted a separate request for exemption from 10 CFR 50.54(m) for VY in Reference 4.

10 CFR 50.54(hh) establishes the requirements for developing, implementing and maintaining procedures and strategies for addressing potential aircraft threats and large area fires or explosions. 10 CFR 50.54(hh)(3) states that this section of the regulation does not apply to nuclear power plants that have submitted the certifications required by 10 CFR 50.82(a).

3.2 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Pursuant to 10CFR50.92, Entergy Nuclear Operations, Inc. (ENO) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed changes would revise and remove certain requirements contained within Section 6.0, Administrative Controls, of the Vermont Yankee Nuclear Power Station (VY) Technical Specifications (TS) and remove the Mitigation Strategy License Condition from the Renewed Facility Operating License (OL). The OL and TS requirements being changed would not be applicable once it has been certified that all fuel has permanently been removed from the VY reactor in accordance with 10 CFR 50.82(a)(1)(ii). Once the certifications for permanent cessation of operations and permanent fuel removal are made, the 10 CFR Part 50 license for VY no longer will authorize operation of the reactor or placement of fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2).

The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

1. <u>Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

   Response: No.

   The proposed amendment would not take effect until VY has permanently ceased operation and entered a permanently defueled condition. The proposed amendment would modify the VY OL and TS by deleting the portions of the OL and TS that are no longer applicable to a permanently defueled facility, while modifying the other sections to correspond to the permanently defueled condition.

   The deletion and modification of provisions of the administrative controls do not directly affect the design of structures, systems, and components (SSCs) necessary for safe storage of irradiated fuel or the methods used for handling and storage of such fuel in the fuel pool. The changes to the administrative controls are administrative in nature and do not affect any accidents applicable to the safe management of irradiated fuel or the permanently shutdown and defueled condition of the reactor. The deletion of the Mitigation Strategy License Condition is also administrative in nature as the sections of the Order requiring implementation of the condition have been rescinded and the controlling regulation in which the mitigation strategies have been codified, 10 CFR 50.54(hh), specifies that these requirements are not applicable in the permanently defueled condition.

   In a permanently defueled condition, the only credible accident is the fuel handling accident.

   The probability of occurrence of previously evaluated accidents is not increased, since extended operation in a defueled condition will be the only operation allowed, and therefore bounded by the existing analyses. Additionally, the occurrence of postulated accidents associated with reactor operation is no longer credible in a permanently defueled reactor. This significantly reduces the scope of applicable accidents.

   Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?</u>

   Response: No.

   The proposed changes have no impact on facility SSCs affecting the safe storage of irradiated fuel, or on the methods of operation of such SSCs, or on the handling and storage of irradiated fuel itself. The administrative removal of an OL condition removal or modifications of the TS that are related only to administration of facility cannot result in different or more adverse failure modes or accidents than previously evaluated because the reactor will be permanently shutdown and defueled and VY will no longer authorized to operate the reactor.

   The proposed deletion of requirements of the VY OL and TS do not affect systems credited in the accident analysis for the fuel handling accident at VY. The proposed OL and TS will continue to require proper control and monitoring of safety significant parameters and activities.

   The proposed amendment does not result in any new mechanisms that could initiate damage to the remaining relevant safety barriers for defueled plants (fuel cladding and spent fuel cooling). Since extended operation in a defueled condition will be the only operation allowed, and therefore bounded by the existing analyses, such a condition does not create the possibility of a new or different kind of accident.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. <u>Does the proposed amendment involve a significant reduction in a margin of safety?</u>

   Response: No.

   Because the 10 CFR Part 50 license for VY will no longer authorize operation of the reactor or emplacement or retention of fuel into the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) are submitted, as specified in 10 CFR 50.82(a)(2), the occurrence of postulated accidents associated with reactor operation is no longer credible. The only remaining credible accident is a fuel handling accident (FHA). The proposed amendment does not adversely affect the inputs or assumptions of any of the design basis analyses that impact the FHA.

   The proposed changes are limited to those portions of the OL and TS that are not related to the safe storage of irradiated fuel. The requirements that are proposed to be revised or deleted from the VY OL and TS are not credited in the existing accident analysis for the remaining applicable postulated accident; and as such, do not contribute to the margin of safety associated with the accident analysis. Postulated DBAs involving the reactor are no longer possible because the reactor will be permanently shutdown and defueled and VY will no longer be authorized to operate the reactor.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, ENO concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.3 PRECEDENT

The proposed changes are consistent with the existing TS Administrative Controls currently in effect for Millstone Nuclear Power Station (DPR-21), which was last substantively revised on March 31, 2001 (Reference 5). The Millstone license amendment that was issued to reflect the permanently shutdown status of the plant on November 9, 1999 (Reference 6), contains TS Administrative Controls similar to those being proposed herein.

The proposed changes are also consistent with the TS Administrative Controls issued to Zion Nuclear Power Station on December 30, 1999 (Reference 7) to reflect the permanently shutdown status of the plant.

3.4 CONCLUSION

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4. ENVIRONMENTAL CONSIDERATIONS

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10CFR51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section 3 of this evaluation, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment does not involve any physical alterations to the plant configuration that could lead to a change in the type or amount of effluent release offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, ENO concludes that the proposed change meets the eligibility criteria for categorical exclusion as set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5. REFERENCES

1. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Notification of Permanent Cessation of Power Operations," BVY 13-079, dated September 23, 2013

2. Letter, Entergy Nuclear Operations, Inc. to USNRC "Request for Approval of Certified Fuel Handler Training Program," BVY 13-095, dated October 31, 2013

3. NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," Revision 4

4. Letter, Entergy Nuclear Operations, Inc. to USNRC "Request for Exemption from 10 CFR 50.54(m)," BVY 13-094, dated October 31, 2013

5. Millstone Nuclear Power Station, Unit 1, Amendment No.109, License No. DPR-21, Date of Issuance March 31, 2001 (ADAMS Accession No. ML010920303)

6. NRC Safety Evaluation for Millstone Power Station Unit 1 in License Amendment 106 to DPR-21, dated November 9, 1999 (ADAMS Accession Nos. ML993330283 and ML993330269)

7. NRC Safety Evaluation for Zion Nuclear Station in License Amendments 180 and 167 (for Units 1 and 2 respectively (License Nos. DPR-39 and DPR-48)), dated December 30, 1999 (ADAMS Accession Nos. ML003672704 and ML003672696)

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Change 307

Markup of the Current Operating License and Technical Specification Pages

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.

Deleted

N. ~~Mitigation Strategy License Condition~~

~~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. 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.

P. The information in the UFSAR supplement, submitted pursuant to 10 CRF 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. 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.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The plant manager shall be responsible for overall unit 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.

shift

C. The shift supervisor shall be responsible for the ~~control room~~ command function. ~~During any absence of the shift supervisor from the control room while the unit is in plant startup or normal operation, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in cold shutdown or refueling with fuel in the reactor, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.~~

6.2 ORGANIZATION

A. Onsite and Offsite Organizations

Organizations shall be established for unit operation and corporate management. These organizations shall include the positions for activities affecting safety of the nuclear ~~power plant~~.

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 unit safe operation and shall have control over those on-site activities necessary for safe ~~operation~~ and maintenance of the ~~plant~~.

storage

3. ~~The site vice president~~ 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 ~~nuclear safety~~.

nuclear fuel

A specified corporate officer

safe management of nuclear fuel

Certified Fuel Handlers

6.2 ORGANIZATION (Cont'd)

4. The individuals who train the ~~operating staff~~, 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 ~~independence from operating pressures~~.

ability to perform their assigned functions

B. Unit Staff

The unit staff organization shall include the following:

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.

1. ~~A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned during Plant Startup and Normal Operation.~~

2. ~~At least one licensed Reactor Operator (RO) or one licensed Senior Reactor Operator (SRO) shall be present in the control room when fuel is in the reactor.~~

3. ~~When the unit is in Plant Startup or Normal Operation, at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, shall be present in the control room.~~

Insert 1

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

4. Shift crew composition shall meet the requirements stipulated herein~~ and in 10 CFR 50.54(m)~~. Shift crew composition may be less than the minimum requirement of ~~10 CFR 50.54(m)(2)(i) and~~ Specification~~s~~ 6.2.B.1 ~~and 6.2.B.8~~ 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 Insert 2

5. An individual qualified in radiation protection procedures shall be present on-site ~~when there is fuel in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position~~. Insert 3

Deleted

6. ~~Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).~~

7. ~~The operations manager or an assistant operations manager shall hold an SRO license~~.

Deleted

8. ~~While the unit is in Plant Startup or Normal Operation an individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operations of the unit. This individual shall meet the qualifications specified by ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Rev. 3, 2000.~~

The shift supervisor shall be a Certified Fuel Handler

6.2 ORGANIZATION (Cont'd)

C. Unit Staff Qualifications

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

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

2. ~~For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed Reactor Operator (RO) are those individuals who, in addition to meeting the requirements of Specification 6.2.C.1, perform the functions described in 10 CFR 50.54(m)~~.

6.3 ~~ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED~~

Deleted

~~Applies to administrative action to be followed in the event a safety limit is exceeded.~~

~~If a safety limit is exceeded, the reactor shall be shutdown immediately.~~

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 of the facility.

B. ~~Refueling~~ Fuel handling operations.

C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components~~, suspected Primary System leaks and abnormal reactivity changes~~.

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 ~~reactor~~ 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:

B. Deleted

Deleted C. ~~Core Operating Limits Report~~

~~The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following:~~

~~1. The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a,~~

~~2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a,~~

~~3. The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a and 3.11.B, and~~

~~4. The Power/Flow Exclusion Region for Specifications 3.6.J.1.a and 3.6.J.1.b.~~

~~The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:~~

~~Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).~~

~~Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (Approved by NRC, SER, dated September 15, 1982).~~

~~Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (Approved by NRC SER, dated September 15, 1982).~~

~~Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1249P, April 1981 (Approved by NRC SER, dated September 27, 1985).~~

~~Report, A. A. F. Ansari, "Methods for the Analysis of Boiling Water Reactors: Steady-State Core Flow Distribution Code (FIBWR)," YAEC-1234, December 1980 (Approved by NRC SER, dated September 15, 1982).~~

~~Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code Qualification and Application," YAEC-1265P, June 1981 (Approved by NRC SER, dated September 27, 1985).~~

~~Report, A. A. F. Ansari and J. T. Cronin, "Methods for the Analysis of Boiling Water Reactors: A System Transient Analysis Model (RETRAN)," YAEC-1233, April 1981. (Approved by NRC SERs, dated November 27, 1981 and September 4, 1984).~~

~~Report, A. A. F. Ansari, K. J. Burns and D. K. Beller, "Methods for the Analysis of Boiling Water Reactors: Transient Critical Power Ratio Analysis (RETRAN-TCPYA01)," YAEC-1299P, March 1982 (Approved by NRC SER, dated September 15, 1982).~~

~~Report, A. S. DiGiovine, et al., "CASMO-3G Validation," YAEC-1363-A, April 1988.~~

~~Report, A. S. DiGiovine, J. P. Gorski, and M. A. Tremblay, "SIMULATE-3 Validation and Verification," YAEC-1659-A, September 1988.~~

~~Report, R. A. Woehlke, et al., "MICBURN-3/CASMO-3/TABLES-3/SIMULATE-3 Benchmarking of Vermont Yankee Cycles 9 through 13," YAEC-1683-A, March 1989.~~

~~Report, J. T. Cronin, "Method for Generation of One-Dimensional Kinetics Data for RETRAN-02," YAEC-1694-A, June 1989.~~

~~Report, V. Chandola, M. P. LeFrancois, and J. D. Robichaud, "Application of One-Dimensional Kinetics to Boiling Water Reactor Transient Analysis Methods," YAEC-1693-A, Revision 1, November 1989.~~

~~Report, L. H. Steves, et. al, "HUXY: A Generalized Multirod Heatup Code with 10CFR50, Appendix K Heatup Option: User's Manual," XN-CC-33(A), Revision 1, dated November 14, 1975 (Approved by NRC SER, dated March 6, 1975).~~

~~Report, "RELAP5YA, A Computer Program for Light-Water Reactor System Thermal-Hydraulic Analysis," YAEC-1300P, October 1982 (Approved by NRC SERs, dated August 25, 1987 and October 21, 1992).~~

~~Report, R. T. Fernandez and H. C. daSilva, Jr., "Vermont Yankee BWR Loss-of-Coolant Accident Licensing Analysis Method," YAEC-1547, June 1986 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from R. W. Capstick (VYNPC) to USNRC, "HUXY Computer Code Information for the Vermont Yankee BWR LOCA Licensing Analysis Method," FVY 87-63, dated June 4, 1987 (Approved by NRC SER, dated February 27, 1991).~~

~~Letter from R. W. Capstick (VYNPC) to USNRC, "Request for Supplemental Safety Evaluation Report Supporting the Use of RELAP5YA for Vermont Yankee Nuclear Power Station," FVY 88-006, dated January 26, 1988 (Approved by NRC SERs, dated February 27, 1991 and October 21, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplementary Information Regarding NRC LOCA Analysis Review Effort," BVY 89-91, dated October 6, 1989 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplementary Information Regarding NRC LOCA Analyses Review Effort," BVY 90-028, dated March 9, 1990 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Second Request for Additional Information on the Use of RELAP5YA," BVY 90-067, dated June 8, 1990 (Approved by NRC SER, dated February 27, 1991).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Request for Additional Information on the Use of RELAP5YA," BVY 90-087, dated August 28, 1990 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Second Request for Additional Information on the Use of RELAP5YA," BVY 91-05, dated January 9, 1991 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Response to Third Request for Additional Information on the Use of RELAP5YA," BVY 91-41, dated April 19, 1991 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplementary Information Regarding the Use of RELAP5YA," BVY 92-12, dated February 7, 1992 (Approved by NRC SER, dated October 21, 1992).~~

~~Letter from R. W. Capstick (VYNPC) to USNRC, "Vermont Yankee LOCA Analysis Method FROSSTEY Fuel Performance Code (FROSSTEY-2)," FVY 87-116, dated December 16, 1987 (Approved by NRC SER, dated September 24, 1992).~~

~~Letter from R. W. Capstick (VYNPC) to USNRC, "Response to NRC Request for Additional Information on the FROSSTEY-2 Fuel Performance Code," BVY 89-65, dated July 14, 1989 (Approved by NRC SER, dated September 24, 1992).~~

~~Letter from R. W. Capstick (VYNPC) to USNRC, "Supplemental Information on the FROSSTEY-2 Fuel Performance Code," BVY 89-74, dated August 4, 1989 (Approved by NRC SER, dated September 24, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Responses to Request for Additional Information on FROSSTEY-2 Fuel Performance Code," BVY 90-045, dated April 19, 1990 (Approved by NRC SER, dated September 24, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplemental Information to VYNPC April 19, 1990 Response Regarding FROSSTEY-2 Fuel Performance Code," BVY 90-054, dated May 10, 1990 (Approved by NRC SER, dated September 24, 1992).~~

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~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "LOCA-Related Responses to Open Issues on FROSSTEY-2 Fuel Performance Code," BVY 92-39, dated March 27, 1992 (Approved by NRC SER, dated September 24, 1992).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "FROSSTEY-2 Fuel Performance Code - Vermont Yankee Response to Remaining Concerns," BVY 92-54, dated May 15, 1992 (Approved by NRC SER, dated September 24, 1992).~~

~~Report, "Loss-of-Coolant Accident Analysis for Vermont Yankee Nuclear Power Station," NEDO-21697, August 1977, as amended (Approved by NRC SER, dated November 30, 1977).~~

~~Report, "General Electric Standard Application for Reactor Fuel (GESTARII)," NEDE-24011-P-A, GE Company Proprietary (the latest NRC-approved version will be listed in the COLR).~~

~~Report, General Electric Nuclear Energy, "BWR Owner's Group Long-Term Solutions Licensing Methodology," NEDO-31960, June 1991 (Approved by NRC SER, dated July 12, 1993).~~

~~Report, General Electric Nuclear Energy, "BWR Owner's Group Long-Term Solutions Licensing Methodology," NEDO-31960, Supplement 1, March 1992 (Approved by NRC SER, dated July 12, 1993).~~

~~Report, N. Fujita, et al., "Method for Power/Flow Exclusion Region Calculation Using the LAPUR5 Computer Code," YAEC-1926-A (Approved by NRC SER, dated November 5, 1996).~~

~~Report, Yankee Atomic Electric Company, "Application of the FIBWR2 Core Hydraulics Code to BWR Reload Analysis," YAEC-1339-A, January 31, 1997.~~

~~The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC.~~

D. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit 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 unit. 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 unit 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. ~~INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT~~

Deleted →

~~A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels will be~~

~~implemented. This program shall include the following:~~

~~1. Provisions establishing preventive maintenance and periodic visual inspection requirements.~~

~~2. System leakage inspections, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are: (1) Residual Heat Removal, (2) Core Spray, (3) Reactor Water Cleanup, (4) HPCI, (5) RCIC, and (6) Sampling Systems.~~

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.

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 ~~review by PORC and approved~~ 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.

Deleted

C. ~~PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM~~

~~A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program," dated September 1995, as modified by the following:~~

- ~~The first Type A test after the April 1995 Type A test shall be performed prior to startup from the April 2010 refuel outage. (This is an exception to Section 9.2.3 of NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR50, Appendix J.")~~

- ~~The leakage contributions from the main steam pathways are excluded from the sum of the leakage rates from Type B and C tests specified in (1) Section III.B of 10CFR50, Appendix J - Option B; (2) Section 6.4.4 of ANSI/ANS 56.8-1994; and (3) Section 10.2 of NEI 94-01, Rev. 0.~~

- ~~The leakage contributions from the main steam pathways are excluded from the overall integrated leakage rate from Type A tests specified in (1) Section III.A of 10CFR50, Appendix J - Option B; (2) Section 3.2 of ANSI/ANS 56.8-1994; and (3) Sections 8.0 and 9.0 of NEI 94-01, Rev. 0.~~

~~The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 44 psig.~~

~~The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.8% of primary containment air weight per day.~~

~~Leakage rate acceptance criteria are:~~

1. ~~Primary containment leakage rate acceptance criterion ≤ 1.0 La.~~

2. ~~The as-left primary containment integrated leakage rate test (Type A test) acceptance criterion is ≤ 0.75 La.~~

3. ~~The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from the main steam pathways) is ≤ 0.6 La, calculated on a maximum pathway basis, prior to entering a mode of operation where primary containment integrity is required.~~

4. ~~The combined local leakage rate test acceptance criterion for Type B and Type C tests (excluding the leakage contributions from the main steam pathways) is ≤ 0.6 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required.~~

~~5. Airlock overall leakage rate acceptance criterion is~~ ≤ ~~0.10 La when tested at~~ ≥ ~~Pa.~~

~~The provision of SR for 4.0.2 for Surveillance Frequency does not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.~~

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 unit 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;

Insert 1

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.

Insert 2

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.

Insert 3

during the movement of fuel and during the movement of loads over fuel

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Change 307

Retyped Operating License and Technical Specification Pages

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.

N. Deleted

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.

P. The information in the UFSAR supplement, submitted pursuant to 10 CRF 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. 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.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The plant manager shall be responsible for overall unit 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 unit operation 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 unit 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.

6.2 ORGANIZATION (Cont'd)

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.

B. Unit Staff

The unit 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

6.2 ORGANIZATION (Cont'd)

C. Unit Staff Qualifications

1. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the Entergy 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 of the facility.

B. Fuel handling operations.

C. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components.

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:

B. Deleted

C. Deleted

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D. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit 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 unit. 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 unit 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.

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 unit 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;