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BVY 14-010

March 28, 2014

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

SUBJECT: Technical Specifications Proposed Change No. 309
Defueled Technical Specifications and Revised License Conditions 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 (ML13273A204)

2. Letter, Entergy Nuclear Operations, Inc. to USNRC "Request for Approval of Certified Fuel Handler Training Program," BVY 13-095, dated October 31, 2013 (ML13325B015)

3. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Technical Specifications Proposed Change No. 307, Revision to Mitigation Strategy License Condition and Technical Specification Administrative Controls for Permanently Defueled Condition," BVY 13-096, dated October 31, 2013 (ML13316A004)

4. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Technical Specifications Proposed Change No. 306, Eliminate Certain ESF Requirements during Movement of Irradiated Fuel," BVY 13-097, dated November 14, 2013 (ML13323A518)

5. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Request for Rescission of Security Orders," BVY 13-105, dated December 18, 2013 (ML13365A007)

6. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Cyber Security Plan Implementation Schedule, Proposed Change No. 308," BVY 13-109, dated December 19, 2013 (ML13358A338)

Dear Sir or Madam:

In accordance with 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) is proposing an amendment to Renewed Facility Operating License (OL) DPR-28 for Vermont Yankee Nuclear Power Station (VY). The proposed amendment would revise the OL and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

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 docketed 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. In support of this condition, the VY OL and associated TS are being proposed for revision to reflect the planned permanently shutdown and defueled condition in accordance with 10 CFR 50.36(c)(6).

Reference 2 was submitted proposing a Certified Fuel Handler training program for NRC approval. Reference 3 proposed changes to eliminate the Mitigation Strategy license condition from the VY OL and changes to the staffing and training requirements for the VY staff contained in Section 6.0, Administrative Controls, of the VY TS to reflect the revised station organization, including Certified Fuel Handlers, which will be implemented once VY is permanently defueled. Reference 3 also proposed additional changes to Section 6.0, Administrative Controls, to delete certain required reports and programs that will no longer be applicable once VY is permanently defueled.

Reference 4 is an amendment request to eliminate certain Engineered Safety Feature requirements during movement of irradiated fuel. Specifically, the changes would eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel or a fuel cask and while performing core alterations. The request also includes a new analysis based on the Fuel Handling Accident for NRC review.

In support of removal of the license condition from the VY OL that requires implementation of actions associated with NRC Order EA-06-137, ENO submitted Reference 5 to request rescission of Order EA-06-137 (Key Radiological Mitigation Strategies).

In Reference 6, ENO proposed a change to the VY Cyber Security Plan Milestone 8 full implementation date. This date change is reflected in the Security Plan license condition in the OL. The six referenced licensing actions complement and support this proposed license amendment.

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, including Bases (TS sections that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 2). Attachment 3 contains the retyped OL and TS pages, including Bases. Bases changes are provided for information only.

ENO requests review and approval of this proposed license amendment on February 1, 2015 with a 60 day implementation period from the effective date of the amendment. This date will allow for VY to complete defueling operations and provide for sufficient decay of the reactor fuel to occur, following an assumed shutdown by the end of December 2014, such that the fuel will no longer be considered recently irradiated, as defined in Reference 4. ENO requests that the approved amendment become effective following NRC approval of References 2 through 6 and docketing 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 March 28, 2014.

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 309

Description and Evaluation of Proposed Changes

1. SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) requests an amendment to Facility Operating License (OL) Number DPR-28 for Vermont Yankee Nuclear Power Station (VY). The proposed amendment would revise the OL and revise the associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the permanently defueled condition of the reactor.

On August 27, 2013, ENO announced that Vermont Yankee Nuclear Power Station (VY) would permanently cease power operations at the end of the current operating cycle. In Reference 1, ENO provided formal notification to the NRC pursuant to 10 CFR 50.82(a)(1)(i).

The proposed changes would revise and remove certain requirements contained within the OL and TS, and remove the requirements that would no longer 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 docketed, the 10 CFR Part 50 license for VY no longer will authorize operation of the reactor or emplacement or retention 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 13 days after final reactor shutdown (assumed to occur no later than the end of December 2014) (Reference 1) to allow for sufficient decay of the reactor fuel such that the fuel will no longer be considered recently irradiated, as defined in Reference 2. The proposed changes to the OL and TS being proposed for deletion or revision are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages and sections, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS table of contents is also accordingly revised.

The existing VY TS contain Limiting Conditions for Operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TS provide an appropriate level of control. However, the majority of the existing TS are only applicable with the reactor in an operational mode. LCOs and associated Surveillance Requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision and incorporation as the PDTS to provide a continuing acceptable level of safety which addresses the reduced scope of postulated design basis accidents associated with a defueled plant.

Pending Licensing Actions under NRC Review

There are several other pending licensing actions currently under NRC review that complement and support the changes proposed by this amendment request. For clarity, the marked-up and retyped OL and TS pages included in Attachments 2 and 3, respectively, reflect the changes proposed by these pending actions. Reference 3 proposed changes to eliminate the Mitigation Strategy license condition from the VY OL and changes to the staffing and training requirements for the VY staff contained in Section 6.0, Administrative Controls, of the VY TS to reflect the revised station organization, including Certified Fuel Handlers, which will be implemented once VY is permanently defueled. Reference 3 also proposed additional changes to Section 6.0, Administrative Controls, to delete certain required reports and programs that will no longer be applicable once VY is permanently defueled. Reference 4 was submitted proposing a Certified Fuel Handler training program for NRC approval.

Reference 2 is an amendment request to eliminate certain Engineered Safety Feature (ESF) requirements during movement of irradiated fuel. Specifically, the changes would eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel or a fuel cask and while performing core alterations. The request also includes a new analysis based on the Fuel Handling Accident (FHA) for NRC review.

In support of removal of the license condition from the VY OL that requires implementation of actions associated with NRC Order EA-06-137, ENO submitted Reference 5 to request rescission of Order EA-06-137 (Key Radiological Mitigation Strategies).

In Reference 6, ENO proposed a change to the VY Cyber Security Plan Milestone 8 full implementation date. This date change is reflected in the Security Plan license condition in the OL.

There are no other pending license amendment requests currently docketed for VY. Therefore, no disposition of other TS changes, as they relate to this license amendment request, is needed.

2. DETAILED DESCRIPTION AND BASIS FOR THE CHANGES

The proposed amendment would modify the VY license and revise VY TS into PDTS to comport with a permanently defueled condition.

General Analysis Applicable to Proposed Change

Section 14 of the VY Updated Final Safety Analysis Report (UFSAR) describes the design basis accident (DBA) and transient scenarios applicable to VY during power operations. During normal power operations, the forced inlet flow of water through the reactor coolant system (RCS) removes the heat from the reactor by generating steam. The steam system, operating at high temperatures and pressures, transfers this heat to the turbine generator. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the reactor coolant system. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems which could affect the reactor core.

However, as a result of the notification of permanent cessation of power operations submitted by ENO pursuant to 10 CFR 50.82(a)(1), and the planned removal of authorization to operate the reactor or to place or retain fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2) once it has been certified that all fuel has been permanently removed from the reactor, most of the DBA scenarios postulated in the UFSAR will no longer be possible. The irradiated fuel will be stored in the spent fuel pool (SFP) and the Independent Spent Fuel Storage Installation (ISFSI) until it is shipped off site in accordance with the schedules to be provided in the Post Shutdown Decommissioning Activities Report (PSDAR) and the updated Irradiated Fuel Management Plan.

A list of the UFSAR Section 14 DBAs is provided in Section 3.1, "Applicable Regulatory Requirements/Criteria," of this submittal. The analyzed accident that remains applicable to VY in the permanently shut down and defueled condition is a FHA. Table 5-3 of the new FHA-based analysis (Reference 2) for VY shows that, following 13 days of decay time after reactor shutdown, the dose consequences are acceptable without relying on structures, systems, and components (SSCs) remaining operable for accident mitigation during and following the event.

10 CFR 50.36, "Technical Specifications," promulgates the regulatory requirements related to the content of Technical Specifications. As detailed in subsequent sections of this proposed amendment, this regulation lists four criteria to define the scope of equipment and parameters that must be included in TS. In a permanently defueled condition, the scope of equipment and parameters that must be included in the VY TS is limited to those needed to address the remaining applicable DBA (i.e., the postulated FHA) so that the consequences of the accident are maintained within acceptable limits.

Fuel Handling Accident Analysis for the Permanently Defueled Condition

The FHA is the only DBA that can occur with the facility in the permanently defueled condition. In Reference 2, ENO provided an FHA-based analysis using Alternate Source Term methodology. The analysis determined the projected dose due to the drop of a fuel assembly onto other fuel assemblies as a function of time after shutdown. The analysis demonstrates that radiological doses at the exclusion area boundary, low population zone and in the control room from a FHA after 13 days following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system. No equipment is required to mitigate the effects of this event beyond the administrative controls described in Reference 2.

Detailed Discussion

The following table identifies each section that is being changed, the proposed changes, and the basis for each change. Changes to the OL are listed first followed by the TS. Proposed changes to the TS Bases are provided for information in Attachment 2 (TS and associated Bases sections that are deleted in their entirety are identified as such, but the associated deleted pages are not included in Attachment 2). Upon approval of this amendment, changes to the Bases will be incorporated in accordance with TS 6.7.E, the TS Bases Control Program.

The proposed changes to the TS are considered a major rewrite which includes a renumbering of pages and sections and deletion of unused placeholders where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS table of contents is revised to reflect the remaining applicable sections and new page numbering. These changes are considered administrative and are shown in the marked-up (Attachment 2) and retyped OL and TS pages (Attachment 3). Since the changes to the TS are considered a major rewrite, revision bars are not used. Note that TS 3/4.9 and 3/4.13 were previously deleted and are not discussed further in this proposed change.

LICENSE FINDING a, CONSTRUCTION	
Current License Finding a. a. Construction of the Vermont Yankee Nuclear Power Station (the facility) has been substantially completed in conformity with the application, as amended, the Provisional Construction Permit No. CPPR-36, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in Title 10 of the *Code of Federal Regulations* (CFR) Chapter 1,: and	Proposed License Finding a. a. Deleted
Basis	
This section is proposed for deletion in its entirety. Decommissioning of VY is not dependent on the regulations that governed construction of the facility.	

LICENSE FINDING b, OPERATION	
Current License Finding b. b. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission; and	Proposed License Finding b. b. The facility will be prohibited from operating the reactor in conformity with the provisions of the Atomic Energy Act of 1954, as amended, and the rules and regulations of the Commission; and
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor.	

LICENSE FINDING c, ACTIVITIES	
Current License Finding c. c. There is reasonable assurance (1) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and	Proposed License Finding c. c. There is reasonable assurance (1) that the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission; and
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor.	

LICENSE FINDING d, FINANCIAL QUALIFICATION	
Current License Finding d. d. Entergy Nuclear Vermont Yankee, LLC is financially qualified and Entergy Nuclear Operations, Inc. is technically and financially	Proposed License Finding d. d. Entergy Nuclear Vermont Yankee, LLC is financially qualified and Entergy Nuclear Operations, Inc. is technically and financially

qualified to engage in the activities authorized by this renewed operating license, in accordance with the rules and regulations of the Commission; and	qualified to engage in the activities authorized by this license, in accordance with the rules and regulations of the Commission; and
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor.	

LICENSE FINDING f, COMMON DEFENSE AND SECURITY	
Current License Finding f. f. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public; and	Proposed License Finding f. f. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public; and
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor.	

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

This section is revised to reflect the change from an operating license to being prohibited from operating the reactor.

LICENSE FINDING h, AGING EFFECTS	
Current License Finding h. h. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.	Proposed License Finding h. h. Actions have been identified and have been or will be taken with respect to: (1) managing the effects of aging on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1) during the period of extended operation, and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3 for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) will prohibit operation of the VY reactor.	

LICENSE CONDITION 2.A, UTILIZATION FACILITY	
Current License Condition 2.A A. Pursuant to Sections 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Entergy Nuclear Vermont Yankee, LLC to possess and use, and Entergy Nuclear Operations, Inc., to possess, use, and	Proposed License Condition 2.A A. Pursuant to Sections 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Entergy Nuclear Vermont Yankee, LLC to possess and use, and Entergy Nuclear Operations, Inc. to possess and use the facility as a utilization facility at the

operate the facility as a utilization facility at the designated location on the Entergy Nuclear Vermont Yankee, LLC site.	designated location on the Entergy Nuclear Vermont Yankee, LLC site.
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor.	

LICENSE CONDITION 2.B, REACTOR FUEL	
Current License Condition 2.B B. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as supplemented and amended.	Proposed License Condition 2.B B. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage as described in the Final Safety Analysis Report, as supplemented and amended.
Basis	
This section is revised to reflect the change from an operating license to being prohibited from operating the reactor and that special nuclear material can no longer be used as reactor fuel. The language regarding receipt and use of special nuclear material as reactor fuel is proposed for deletion (and referring to use of reactor fuel in the past tense).	

LICENSE CONDITION 2.C, SPECIAL NUCLEAR MATERIAL	
Current License Condition 2.C C. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for calibration of reactor instrumentation and radiation monitoring equipment, and as fission detectors in amounts as required.	Proposed License Condition 2.C C. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in radiation monitoring equipment, and as fission detectors in amounts as required.

Basis	
This section is revised regarding receipt and use of sealed neutron sources for reactor startup. This license condition is revised to reflect authorization only for continued possession of those sources used for reactor startups. The Part 50 license will no longer authorize operation of the facility and this condition will no longer authorize receipt and use of sources used for reactor startup or for calibration of reactor instrumentation.	

LICENSE CONDITION 3.A, MAXIMUM POWER LEVEL	
Current License Condition 3.A Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1912 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.	Proposed License Condition 3.A Deleted
Basis	
This section is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the VY reactor since the certifications described therein will have been submitted.	

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

affect the ability to achieve and maintain safe shutdown in the event of a fire.	

Basis

This section is proposed for deletion to reflect the permanently defueled condition of the facility. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, the fire protection program will be revised to take into account the decommissioning plant conditions and activities. VY will continue to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment.

License Condition 3.F**,** which is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire, will no longer be applicable at VY. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shutdown and defueled plant is not needed.

LICENSE CONDITION 3.G, SECURITY PLAN

Changes to License Condition 3.G, Security Plan, are under NRC review. Reference: Letter, Entergy Nuclear Operations, Inc., to NRC, "Cyber Security Plan Implementation Schedule Proposed Change No. 308," BVY 13-109, dated December 19, 2013 (Reference 6).

A revised security plan and any identified exemption requests to 10 CFR 73.55 to reflect the permanently shutdown condition will be submitted under separate cover.

LICENSE CONDITION 3.K, MINIMUM CRITICAL POWER RATIO

Current License Condition 3.K	Proposed License Condition 3.K
K. Minimum Critical Power Ratio When operating at thermal power greater than 1593 megawatts thermal, the safety limit minimum critical power ratio (SLMCPR) shall be established by adding 0.02 to the cycle-specific SLMCPR value calculated using the NRC-approved methodologies documented in General Electric Licensing Topical Report NEDE-24011-P-A, “General Electric Standard	K. Deleted

Application for Reactor Fuel," as amended, and documented in the Core Operating Limits Report.	

Basis

This section is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the VY reactor since the certifications required by 10 CFR 50.82(a)(1) will have been submitted. Therefore, a license condition for minimum critical power ratio is no longer needed.

LICENSE CONDITION 3.L, TRANSIENT TESTING

Current License Condition 3.L	Proposed License Condition 3.L
L. Transient Testing 1. During the extended power uprate (EPU) power ascension test program and prior to exceeding 168 hours of plant operation at the nominal full EPU reactor power level, with feedwater and condensate flow rates stabilized at approximately the EPU full power level, Entergy Nuclear Operations, Inc. shall confirm through performance of transient testing that the loss of one condensate pump will not result in a complete loss of reactor feedwater. 2. Within 30 days at nominal full-power operation following successful performance of the test in (1) above, through performance of additional transient testing and/or analysis of the results of the testing conducted in (1) above, confirm that the loss of one reactor feedwater pump will not result in a reactor trip.	L. Deleted

Basis

This section is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. This actions prescribed by this license condition have been completed. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the VY reactor at EPU levels since the certifications required by 10 CFR 50.82(a)(1) will have been submitted. Therefore, this license condition for transient testing is no longer needed.

LICENSE CONDITION 3.M, POTENTIAL ADVERSE FLOW EFFECTS	
<u>Current License Condition 3.M</u> M. Potential Adverse Flow Effects This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer). 1. The following requirements are placed on operation of the facility above the original licensed thermal power (OLTP) level of 1593 megawatts thermal (MWt): a. Entergy Nuclear Operations, Inc. shall monitor hourly the 32 main steam line (MSL) strain gages during power ascension above 1593 MWt for increasing pressure fluctuations in the steam lines. b. Entergy Nuclear Operations, Inc. shall hold the facility for 24 hours at 105%, 110%, and 115% of OLTP to collect data from the 32 MSL strain gages required by Condition M.1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall provide the evaluation to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission. c. If any frequency peak from the MSL strain gage data exceeds the limit curve established by Entergy Nuclear Operations, Inc. and submitted to the NRC staff prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall return the facility to a power level at which the limit curve is not exceeded.	<u>Proposed License Condition 3.M</u> M. Deleted

Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power. d. In addition to evaluating the MSL strain gage data, Entergy Nuclear Operations, Inc. shall monitor reactor pressure vessel water level instrumentation or MSL piping accelerometers on an hourly basis during power ascension above OLTP. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data, Entergy Nuclear Operations, Inc. shall stop power ascension, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power. e. Following start-up testing, Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis and provide that resolution to the NRC staff by facsimile or electronic transmission to the NRC project manager. If the uncertainties are not resolved within 90 days of issuance of the license amendment authorizing operation at 1912 MWt, Entergy Nuclear Operations, Inc. shall return the facility to OLTP. 2. As described in Entergy Nuclear Operations, Inc. letter BVY 05-084 dated September 14, 2005, Entergy Nuclear Operations, Inc. shall implement the following actions: a. Prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall install 32 additional strain gages on the main steam piping and shall enhance the data acquisition system in order to reduce the measurement uncertainty associated with	

the acoustic circuit model (ACM). b. In the event that acoustic signals are identified that challenge the limit curve during power ascension above OLTP, Entergy Nuclear Operations, Inc. shall evaluate dryer loads and re-establish the limit curve based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency. c. After reaching 120% of OLTP, Entergy Nuclear Operations, Inc. shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the steam dryer monitoring plan (SDMP) limit curve with the updated ACM load definition and revised instrument uncertainty, which will be provided to the NRC staff. d. During power ascension above OLTP, if an engineering evaluation is required in accordance with the SDMP, Entergy Nuclear Operations, Inc. shall perform the structural analysis to address frequency uncertainties up to ±10% and assure that peak responses that fall within this uncertainty band are addressed. e. Entergy Nuclear Operations, Inc. shall revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with General Electric Services Information Letter 644, Revision 1; and to identify the NRC Project Manager for the facility as the point of contact for providing SDMP information during power ascension. f. Entergy Nuclear Operations, Inc. shall submit the final extended power uprate (EPU) steam dryer load definition for the facility to the NRC upon completion of the power ascension test program. g. Entergy Nuclear Operations, Inc. shall submit the flow-induced vibration related	

portions of the EPU startup test procedure to the NRC, including methodology for updating the limit curve, prior to initial power ascension above OLTP. 3. Entergy Nuclear Operations, Inc. shall prepare the EPU startup test procedure to include the (a) stress limit curve to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria are not satisfied; and (i) verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above OLTP. Entergy Nuclear Operations, Inc. shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above OLTP. 4. When operating above OLTP, the operating limits, required actions, and surveillances specified in the SDMP shall be met. The following key attributes of the SDMP shall not be made less restrictive without prior NRC approval: a. During initial power ascension testing above OLTP, each test plateau increment shall be approximately 80 MWt; b. Level 1 performance criteria; and c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria. Changes to other aspects of the SDMP may be made in accordance with the guidance of NEI 99-04. 5. During each of the three scheduled refueling	

outages (beginning with the spring 2007 refueling outage), a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer, including flaws left "as is" and modifications. 6. The results of the visual inspections of the steam dryer conducted during the three scheduled refueling outages (beginning with the spring 2007 refueling outage) shall be reported to the NRC staff within 60 days following startup from the respective refueling outage. The results of the SDMP shall be submitted to the NRC staff in a report within 60 days following the completion of all EPU power ascension testing. 7. The requirements of paragraph 4 above for meeting the SDMP shall be implemented upon issuance of the EPU license amendment and shall continue until the completion of one full operating cycle at EPU. If an unacceptable structural flaw (due to fatigue) is detected during the subsequent visual inspection of the steam dryer, the requirements of paragraph 4 shall extend another full operating cycle until the visual inspection standard of no new flaws/flaw growth based on visual inspection is satisfied. 8. This license condition shall expire upon satisfaction of the requirements in paragraphs 5, 6, and 7 provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is due to fatigue.	

Basis

This section is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the VY reactor at EPU levels since the certifications required by 10 CFR 50.82(a)(1) will have been submitted. Additionally, the steam dryer performs the function of removing moisture from the wet steam exiting the steam separators during reactor operation and serves no function in the shutdown condition. Therefore, this license condition is no longer needed.

LICENSE CONDITION 3.N, MITIGATION STRATEGY LICENSE CONDITION

Changes to License Condition 3.N, Mitigation Strategy License Condition, are under NRC review. Reference: Letter, Entergy Nuclear Operations, Inc., to NRC, "Technical Specifications Proposed Change No. 307 Revision to Mitigation Strategy License Condition and Technical Specification Administrative Controls for Permanently Defueled Condition," BVY 13-096, dated October 31, 2013 (ML13316A004) (Reference 3).

LICENSE CONDITION 3.O, RADIOLOGICAL MITIGATION STRATEGIES

Current License Condition 3.O	Proposed License Condition 3.O
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.	O. Deleted

Basis

The NRC issued Order EA-06-137 (Key Radiological Mitigation Strategies) on June 20, 2006, requiring each licensee to amend its site security plan, safeguards contingency plan, guard training and qualification plan, and emergency plan as appropriate to address the key radiological protection mitigation strategies identified for its facilities.

This section is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. The design basis accident (Loss of Coolant Accident) that this order was based on is not a postulated accident that can occur in the permanently defueled condition, rendering the actions required by the order and this license condition unnecessary.

In support of removal of this license condition from the VY OL, ENO submitted Reference 5 to request rescission of Order EA-06-137.

LICENSE CONDITION 3.R, VESSEL AND INTERNALS INTEGRATED SURVEILLANCE PROGRAM

Current License Condition 3.R	Proposed License Condition 3.R
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	R. Deleted

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.	

Basis

10 CFR 50 Appendix H requires that the design of the reactor vessel surveillance capsule program and withdrawal schedule must meet the requirements in the version of American Society for Testing and Materials (ASTM) Standard Practice E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor pressure vessel (RPV) was purchased. The rule also requires the licensee to perform capsule testing and to report the test results in accordance with the requirements in ASTM Standard Practice E 185-82 to the extent practicable for the configuration of the test specimens in the RPV surveillance capsules.

The requirements in Appendix H are only applicable to nuclear plants that are performing power operations in the reactor critical operating mode because: (a) this is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process, and (b) the requirements are set in place to provide assurance that the RPV will maintain adequate levels of fracture toughness throughout the operating life of the reactor.

Continued implementation of the applicable surveillance capsule testing and reporting requirements are no longer necessary for VY because: (a) ENO has decided to cease power operations of VY, and (b) from a fracture toughness perspective, the VY RPV will cease to be exposed to further irradiation by high energy neutrons or subjected to any high thermal stress environments, as induced by operating the RCS at an elevated temperature.

The physical and radiological control of the remaining surveillance capsules that are located in the RPV will be managed in accordance with the applicable radiological control requirements of 10 CFR Part 20 and with any applicable security or physical protection requirements for components in either 10 CFR Part 37 or 10 CFR Part 73. Therefore, the removal, testing, reporting, and storage requirements for reactor vessel surveillance capsules and their test specimens do not need to be implemented further once VY permanently ceases power operations because there will no longer be any need to remove the remaining surveillance capsules from the RPV or perform material testing of the test specimens in those capsules. As such, deletion of this license condition is appropriate. Any corresponding commitments in the VY UFSAR will also be deleted under the provisions of 10 CFR 50.59 upon NRC approval.

LICENSE CONDITION 3.S, STEAM DRYER MONITORING AND INSPECTION PROGRAM	
<u>Current License Condition 3.S</u> S. Steam Dryer License Condition Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. shall perform and implement the continuous parameter monitoring, moisture content monitoring, and visual inspections of the steam dryer described in BWRVIP-139-A, Final Report, dated July 2009, with the additional requirements included in the license condition specified herein. 1. The licensee shall inspect the steam dryer during refueling outage (RFO) 31. If the steam dryer inspection results from RFO 31 demonstrate that (a) any new cracking due to either fatigue or intergranular stress-corrosion cracking (IGSCC) is not significant (i.e., is evaluated to be acceptable without repair), and (b) all previously identified indications (including the new indication identified in RFO 27) show no crack growth requiring repair, the licensee shall initiate a program to perform steam dryer inspections at least every third refueling outage. 2. If the steam dryer inspection during RFO 31 or any subsequent steam dryer inspection reveals new cracking due to either fatigue or IGSCC that requires repair, or any growth of existing cracking that requires repair, the licensee shall inspect the steam dryer during each refueling outage. 3. The steam dryer inspection during RFO 31 and all future steam dryer inspections shall meet the guidelines in BWRVIP-139-A with the exception of the inspection interval specified in BWRVIP-139-A. The results of the steam dryer inspection beginning with RFO 31 and continuing for all future inspections shall be submitted to the NRC within 6 months of each steam dryer inspection completion. 4. This license condition shall continue for the full term of the period of extended	<u>Proposed License Condition 3.S</u> S. Deleted

operation unless duly amended.	

Basis

The steam dryer performs the function of removing moisture from the wet steam exiting the steam separators during reactor operation and serves no function in the shutdown condition. This section is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the VY reactor since the certifications required by 10 CFR 50.82(a)(1) will have been submitted. Therefore, a license condition for the steam dryer monitoring and inspection program is no longer needed.

LICENSE SECTION 4, EXPIRATION OF LICENSE

Current License Section 4.	Proposed License Section 4.
4. This renewed operating license is effective as of the date of issuance and shall expire at midnight on March 21, 2032.	4. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

Basis

This section is revised to reflect the permanently defueled condition of the facility. Once VY has permanently ceased operation and certified that fuel has been removed from the reactor, 10 CFR 50.82(a)(2) prohibits operation of the VY reactor since the certifications required by 10 CFR 50.82(a)(1) will have been submitted. This license section is being revised in accordance with 10 CFR 50.51(b) in that the license authorizes ownership and possession by ENO until the Commission notifies the licensee in writing that the license is terminated.

TS SECTION 1.0, DEFINITIONS

1.0 Definitions

TS 1.0, "Definitions," provides defined terms that are applicable throughout the TS and TS Bases. The following definitions are being proposed for deletion because they have no relevance to and therefore no longer apply to the permanently defueled plant status. A new definition for Certified Fuel Handler is also proposed for addition to the PDTS.

Definitions Being Deleted	**Basis**

Reportable Occurrence - The equivalent of a reportable event which shall be any of the conditions specified in Section 50.73 to 10 CFR Part 50.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. Reportable occurrences are defined and codified in their applicable regulations (e.g. 10 CFR 50.72 and 10 CFR 50.73) and, therefore, the definition need not be repeated in the PDTS.
Alteration of the Reactor Core - The act of moving any component affecting reactivity within the reactor vessel in the region above the core support plate, below the upper grid and within the shroud. Normal movement of control rods or neutron detectors, or the replacement of neutron detectors is not defined as a core alteration.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term has no meaning when there is no fuel in the reactor core.
Hot Standby - Hot standby means operation with the reactor critical and the main steam line isolation valves closed.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term has no meaning when there is no reactor core.
Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time as specified is not part of the routine instrument calibration but will be checked once per operating cycle.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accident that remains credible (the FHA).
Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accident that remains credible.
Instrument Functional Test - An instrument functional test shall be: 1. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accident that remains credible.

and/or trip functions. 2. Bistable channels - the injection of a signal into the sensor to verify the operability including alarm and/or trip functions.	
<u>Logic System Functional Test</u> - A logic system functional test shall be a test of all logic components required for operability of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify operability. The logic system functional test may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There are no logic systems credited in the analysis of the accident that remains credible.
<u>Minimum Critical Power Ratio</u> – The minimum critical power ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the appropriate NRC-approved critical power correlation to the actual assembly operating power.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This term applies only to a critical reactor.
<u>Mode</u> - The reactor mode is that which is established by the mode-selector-switch.	This definition is not proposed for inclusion in the PDTS since operating modes are not used in any PDTS specification. Modes are defined for operating or refueling conditions. This term does not apply to a facility in the permanently defueled condition.
<u>Operating Cycle</u> - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.	This definition is not proposed for inclusion in the PDTS since this term not used in any PDTS specification. There will be no more refueling outages once VY has certified that fuel has been permanently removed from the reactor.
<u>Primary Containment Integrity</u> - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied: 1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment, which are not required to be open during accident conditions, are closed. Such valves may be opened intermittently under administrative controls.	This definition is not proposed for inclusion in the PDTS since this term not used in any PDTS specification. In the permanently defueled condition there will be no DBAs for which primary containment integrity will be required to mitigate the consequences.

2. At least one door in each airlock is closed and sealed. 3. All automatic containment isolation valves are operable or deactivated in the isolated position. 4. All blind flanges and manways are closed.	
<u>Protective Instrumentation Definitions</u> 1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel. 2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems. 3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level. 4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. There is no instrumentation credited in the analysis of the accident that remains credible.
<u>Rated Neutron Flux</u> - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1912 thermal megawatts.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This definition only applies to an operating reactor core.
<u>Rated Thermal Power</u> - Rated thermal power means a steady state power level of 1912 thermal megawatts.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This definition only applies to an operating reactor core.
<u>Reactor Power Operation</u> - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This definition only applies to an operating reactor core.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser vacuum trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable. 2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.	
<u>Reactor Vessel Pressure</u> – Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This definition only applies to an operating reactor.
<u>Refueling Outage</u> - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.	This definition is not proposed for inclusion in the PDTS since this term not used in any PDTS specification. There will be no more refueling outages once VY has certified that fuel has been permanently removed from the reactor.
<u>Shutdown</u> - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized. 1. Hot Shutdown means conditions as above with reactor coolant temperature greater that 212°F.	This definition is not proposed for inclusion in the PDTS since operating modes are not used in any PDTS specification. This definition is meant to differentiate between shutdown modes and operating reactor modes. This term does not apply to a facility in the permanently defueled condition.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F. 3. Shutdown means conditions as above such that the effective multiplication factor (Keff) of the core shall be less than 0.99.	
<u>Transition Boiling</u> - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. This definition only applies to an operating reactor.
<u>Surveillance Frequency</u> – Relocated to Specifications 4.0.2 and 4.0.3.	This definition is not proposed for inclusion in the PDTS since the term is a redundant to the information provided in Specifications 4.0.2 and 4.0.3.
<u>Surveillance Interval</u> - Relocated to Specification 4.0.1.	This definition is not proposed for inclusion in the PDTS since the term is a redundant to the information provided in Specification 4.0.1.
<u>Source Check</u> - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.	This definition is not proposed for inclusion in the PDTS since this term not used in any PDTS specification. There is no radiation detection instrumentation proposed for inclusion in the PDTS that will require a source check.
<u>Dose Equivalent I-131</u> - The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988; FGR 12, “External Exposure to Radionuclides In Air, Water, and Soil,” 1993; or NRC Regulatory Guide 1.109, Revision 1, October 1977.	This definition is not proposed for inclusion in the PDTS since this term not used in any PDTS specification. This term is used to express dose from a mixture of iodine isotopes created in an operating core and contained in plant primary or secondary coolant. The value of Dose Equivalent I-131 is used for dose analysis of accidents involving the primary and secondary coolant releases. Those accident conditions will no longer apply to the permanently shutdown and defueled plant.
<u>Core Operating Limits Report</u> - The Core Operating Limits Report is the unit-specific	This definition is not proposed for inclusion in the PDTS since the term is no longer used and

document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.C. Plant operation within these operating limits is addressed in individual specifications.	the program that required the COLR has been proposed for elimination from the PDTS in Reference 3.
Reactor Protection System (RPS) Response Time - RPS Response Time shall be the time from the opening of the sensor contact up to and including the opening of the scram solenoid relay.	This definition is not proposed for inclusion in the PDTS since the term is not used in any PDTS specification. The RPS will have no function in the permanently shutdown plant.
Definition Being Added	**Basis**
Certified Fuel Handler - A Certified Fuel Handler is an individual who complies with provisions of the Certified Fuel Handler training program.	The definition of the term Certified Fuel Handler is being added to ensure consistent understanding and application. Further discussion regarding Certified Fuel Handlers is included in References 3 and 4.

TS SECTIONS 1/2.1 AND 1/2.2, SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING	
Current TS 1/2.1 1.1 and 2.1 FUEL CLADDING INTEGRITY In summary the fuel cladding integrity Safety Limit (SL) is set such that no significant fuel damage is calculated to occur if the limit is not violated. Current TS 1/2.2 1.2 and 2.2 REACTOR COOLANT SYSTEM In summary the reactor coolant system SL establishes a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.	Proposed TS 1/2.1 Deleted Proposed TS 1/2.2 Deleted
Basis	
All TS in Sections 1/2.1 and 1/2.2, Safety Limit and Limiting Safety System Setting, are being	

proposed for deletion. The safety limits and limiting safety system settings do not apply to a reactor that is permanently defueled.

The restrictions promulgated in TS 1.1 and 2.1 prevent overheating of the fuel and cladding, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. TS 1.1 and 2.1 are applicable in Modes RUN and STARTUP/HOT STANDBY.

TS 1.2 and 2.2 promulgate requirements on parameters to protect the integrity of the reactor coolant system (RCS) against overpressure. TS 1.2 and 2.2 are applicable in all Modes.

Summary

The above TS contain limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the facility when the reactor is in all Modes. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific process variables, are no longer applicable. Based on the above, the proposed deletion of TS related to these safety limits and limiting safety system settings is acceptable. These changes 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.

TS SECTION 3.0, LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

Current LCO 3.0.8, Inoperability of Snubbers	Proposed LCO 3.0.8, Inoperability of Snubbers
LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not being capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and as such, are appropriate for control by the licensee.	Deleted

Basis

TS 3.0, "Limiting Condition for Operation (LCO) Applicability," consists of LCO 3.0.1 through LCO 3.0.8. LCO 3.0.1 through LCO 3.0.8 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. LCO 3.0.1 through LCO 3.0.7 are unused placeholders and proposed for deletion in their entirety.

LCO 3.0.8 – LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s).LCO 3.0.8 is being proposed for deletion in its entirety.

Summary

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, all systems associated with snubbers are no longer required to be operable. As such, the allowance provided by LCO 3.0.8 is no longer needed. Therefore, the proposed deletion of LCO 3.0.8 is acceptable.

TS SECTION, 4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

Current	Proposed
Current SR 4.0.1 SRs shall be met during the modes or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified frequency shall be failure to meet the LCO except as provided in SR 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.	Proposed SR 4.0.1 SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified frequency shall be failure to meet the LCO except as provided in SR 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
Current SR 4.0.2 Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.	Proposed SR 4.0.2 Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%.
Current SR 4.0.3	Proposed SR 4.0.3

If it is discovered that a surveillance was not performed within its specified frequency, declaring applicable Limiting Conditions for Operation (LCOs) not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the surveillance is not performed within the delay period, applicable LCOs must immediately be declared not met, and applicable LCOs must be entered. When the surveillance is performed within the delay period and the surveillance is not met (i.e., acceptance criteria are not satisfied), applicable LCOs must immediately be declared not met, and applicable LCOs must be entered.	No change

Basis

SR 4.0.1 through SR 4.0.3 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 4.0.1 establishes the requirement that SRs must be met during the applicable Modes or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits.

SR 4.0.1 is being retained in the PDTS with a proposed revision. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the reference to operating modes is no longer relevant and is therefore being deleted from the surveillance requirement.

SR 4.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per... " interval.

SR 4.0.2 is being retained in the PDTS with a proposed revision. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the reference to operating cycle is no longer relevant and is therefore being deleted from the surveillance requirement.

SR 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance requirement has not been completed within the specified Frequency.

SR 4.0.3 is being retained in the PDTS unchanged.

TS SECTION 3/4.1, REACTOR PROTECTION SYSTEM

Current TS 3/4.1	Proposed TS 3/4.1
Reactor Protection System (RPS)	Deleted

Basis

Existing TS Section 3/4.1 contains LCOs and SRs for the RPS and is proposed for deletion in its entirety. The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary (RCPB) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually. The RPS is required to be operable in RUN, STARTUP/HOT STANDBY and Refuel with reactor coolant temperature > 212°F, and in Refuel with reactor coolant temperature < 212°F and any control rod withdrawn from a core cell containing one or more fuel assemblies.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. Once VY dockets 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 and will no longer be applicable once VY is in the permanently defueled condition. Based on the above, the proposed deletion of TS Section 3/4.1 in its entirety is acceptable.

TS SECTION 3/4.2, PROTECTIVE INSTRUMENT SYSTEMS

Current TS 3/4.2	Proposed TS 3/4.2
A. Emergency Core Cooling System (ECCS)	A. Deleted
B. Primary Containment Isolation	B. Deleted
C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	C. Deleted
D. Deleted	D. Deleted
E. Control Rod Block Actuation	E. Deleted
F. Mechanical Vacuum Pump Isolation	F. Deleted

Instrumentation	
G. Post-Accident Monitoring Instrumentation	G. Deleted
H. Deleted	H. Deleted
I. Recirculation Pump Trip Instrumentation	I. Deleted
J. Deleted	J. Deleted
K. Degraded Grid Protective System	K. Deleted
L. Reactor Core Isolation Cooling (RCIC) System Actuation	L. Deleted

Basis

Existing TS Section 3/4.2, "Protective Instrument Systems," contains LCOs and SRs to assure and verify operability of protective instrumentation systems. Once VY dockets 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). Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, protective instruments will not be required and these LCOs (and associated SRs) will not apply in a defueled condition. Therefore TS 3/4.2 is proposed for deletion in its entirety.

<u>Emergency Core Cooling Systems (ECCS)</u>

TS 3/4.2.A specifies the operability requirements for ECCS protective instrumentation. As discussed in 10 CFR 50.46(a)(1)(i), the requirement to have an ECCS does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted. Therefore, the need for an associated protective instrumentation system is not required and this specification may be deleted.

<u>Primary Containment Isolation</u>

TS 3/4.2.B specifies the operability requirements for the primary containment isolation instrumentation, which automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. The FHA will be the only credible DBA possible in the permanently defueled condition. The FHA analysis does not rely on primary containment to mitigate the consequences of the FHA. Therefore, this specification may be deleted.

<u>Reactor Building Ventilation and Standby Gas Treatment System</u>

TS 3/4.2.C specifies the operability requirements for the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation. This instrumentation automatically initiates closure of the Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs) and starts the Standby Gas Treatment (SGT) System following DBAs requiring isolation of secondary containment.

Consistent with the changes proposed to footnotes (c) and (d) for Trip Functions 3 and 4 in TS Table 3.2.3 in Reference 2, this specification will no longer be needed once all fuel has been moved to the spent fuel pool and the 13 days have elapsed from the time of plant shutdown. In support of these changes, ENO has completed an FHA-based analysis using the guidelines detailed in Regulatory Guide 1.183. The analysis demonstrates that radiological doses at the exclusion area boundary, low population zone and in the control room from a FHA after 13 days following shutdown are within allowable limits without crediting secondary containment operability and operation of the standby gas treatment system.

Control Rod Block Actuation

TS 3/4.2.E specifies operability requirements for control rod block instrumentation. As discussed in the basis for removal of TS 3/4.3, this specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted because there will no longer be a need for control rods to control core reactivity.

Mechanical Vacuum Pump Isolation Instrumentation

TS 3/4.2.F specifies the operability requirements for the mechanical vacuum pump isolation instrumentation. This instrumentation initiates an isolation of the mechanical vacuum pump following events in which main steam radiation monitors exceed a predetermined value. Tripping and isolating the mechanical vacuum pumps limits control room and offsite doses in the event of a control rod drop accident (CRDA). The CRDA will not be a credible DBA that can occur in the permanently defueled condition. Therefore, this specification may be deleted.

Post-Accident Monitoring Instrumentation

TS 3/4.2.G specifies the operability requirements for the post-accident monitoring (PAM) instrumentation. The purpose of this instrumentation is to display, in the control room, plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs. The parameters monitored by this instrumentation include Drywell Atmospheric Temperature, Drywell Pressure, Torus Pressure, Torus Water Level, Torus Water Temperature, Reactor Pressure, Reactor Vessel Water Level, Torus Air Temperature and Containment High Range Radiation Monitor. None of these parameters are used to provide information to operators during a FHA, which will be the only credible DBA in the permanently defueled condition. Therefore, this specification may be deleted.

Recirculation Pump Trip Instrumentation

TS 3/4.2.I specifies the operability requirements for the Recirculation Pump Trip instrumentation. The Anticipated Transient Without Scram (ATWS) Prevention/Mitigation System initiates a Recirculation Pump Trip (RPT), adding negative reactivity, following events in which a scram does not but should occur, to lessen the effects of an ATWS event. An ATWS will not be a credible event once the plant is in the permanently defueled condition. Therefore, this specification may be deleted.

Degraded Grid Protective System

TS 3/4.K specifies the operability requirements for the Degraded Grid Protective System instrumentation. This instrumentation monitors the 4.16 kV emergency buses. If the monitors determine that insufficient voltage is available and an ECCS initiation signal is present, the buses are disconnected from the offsite power sources and connected to the onsite EDG power sources. As discussed in the basis for removal of TS 3/4.5 and 3/4.10, both ECCS and EDGs will no longer be required once the plant is in the permanently defueled condition. Therefore, this specification will no longer be necessary and may be deleted.

Reactor Core Isolation Cooling (RCIC) System Actuation

TS 3/4.2.L specifies the operability requirements for the RCIC System instrumentation. As discussed in the basis for removal of TS 3/4.5, once VY is in the permanently defueled condition, the conditions for which RCIC is required to operate will no longer be possible. Therefore, this specification will no longer be necessary and may be deleted.

Summary

The protection and monitoring functions of the various instrumentation systems have been designed to ensure safe operation of the reactor and to mitigate the effects of reactor related design basis accidents. Once VY dockets 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 and will no longer be applicable once VY is in the permanently defueled condition. Based on the above, the proposed deletion of TS Section 3/4.2 in its entirety is acceptable.

TS SECTION 3/4.3, CONTROL ROD SYSTEMS	
Current TS 3/4.3	Proposed TS 3/4.3
A. Reactivity Limitations	A. Deleted
B. Control Rods	B. Deleted
C. Scram Insertion Times	C. Deleted
D. Control Rod Accumulators	D. Deleted
E. Reactivity Anomalies	E. Deleted
F. Scram Discharge Volume Vent and Drain Valves	F. Deleted
Basis	

Existing TS Section 3/4.3, "Control Rod Systems," contains LCOs and SRs that ensure the ability of the control rods to control core reactivity. Once VY dockets 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). Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, control rods will not be required and these LCOs (and associated SRs) will not apply in a defueled condition. Therefore TS Section 3/4.3 is proposed for deletion in its entirety.

Reactivity Limitations

TS 3/4.3.A specifies the reactivity limitations during core loading and operation with inoperable control rods. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Control Rods

TS 3/4.3.B specifies the operability requirements for the control rods. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Scram Insertion Times

TS 3/4.3.C specifies the control rod scram insertion times. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Control Rod Accumulators

TS 3/4.3.D specifies the operability requirements for the control rod scram accumulators. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Reactivity Anomalies

TS 3/4.3.E specifies the limits for the reactivity difference between the actual and expected critical control rod configuration while the reactor is operating. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Scram Discharge Volume Vent and Drain Valves

TS 3/4.3.F specifies the operability requirements for the Scram Discharge Volume vent and drain valves while the reactor is in the STARTUP and RUN modes. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Summary

The Control Rod system is only needed to support operation of the reactor. The VY 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. Therefore, this Specification will not be needed and will no longer be applicable once VY is in the permanently defueled condition. Based on the above, the proposed deletion of TS Section 3/4.3 in its entirety is acceptable.

TS SECTION 3/4.4, REACTOR STANDBY LIQUID CONTROL SYSTEM	
Current TS 3/4.4 A. Normal Operation	Proposed TS 3/4.4 A. Deleted

B. Operation With Inoperable Components C. Standby Liquid Control System Tank – Borated Solution	B. Deleted C. Deleted

Basis

Existing TS Section 3/4.4, "Reactor Standby Liquid Control System," contains LCOs and SRs that provide assurance of availability and operability of the Standby Liquid Control (SLC) system as a means of providing an independent reactivity control mechanism and is proposed for deletion in its entirety. 10 CFR 50.62(c)(4) requires each boiling water reactor to have an SLC system. As discussed in 10 CFR 50.62(a), the requirements of 10 CFR 50.62 do not apply to nuclear power reactor facilities for which the certifications required under 10 CFR 50.82(a)(1) have been submitted. Because VY will be submitting the required certifications, the LCOs (and associated SRs) that do not apply in a defueled condition are being proposed for deletion.

Normal Operation

TS 3/4.4.A specifies the requirements for normal operation of the SLC system. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Operation With Inoperable Components

TS 3/4.4.B specifies the requirements for operation with inoperable SLC system components. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Standby Liquid Control System Tank – Borated Solution

TS 3/4.4.C specifies the minimum concentrations of sodium pentaborate solution in the SLC system tank. This specification will not be required once the certifications required under 10 CFR 50.82(a)(1) have been submitted.

Summary

Once VY dockets the certifications required by 10 CFR 50.82(a)(1), 10 CFR 50.62 will no longer be applicable. Therefore, the SLC system and associated TS will not be needed and will no longer be applicable once VY is in the permanently defueled condition. Based on the above, the proposed deletion of TS Section 3/4.4 in its entirety is acceptable.

TS SECTION 3/4.5, CORE AND CONTAINMENT COOLING SYSTEMS

Current TS 3/4.5	Proposed TS 3/4.5
A. Core Spray and Low Pressure Coolant Injection	A. Deleted

B. Containment Spray Cooling Capability	B. Deleted
C. Residual Heat Removal (RHR) Service Water System	C. Deleted
D. Station Service Water and Alternate Cooling Tower Systems	D. Deleted
E. High Pressure Cooling Injection (HPCI) System	E. Deleted
F. Automatic Depressurization System	F. Deleted
G. Reactor Core Isolation Cooling System (RCIC)	G. Deleted
H. Minimum Core and Containment Cooling System Availability	H. Deleted
I. Maintenance of Filled Discharge Pipe	I. Deleted

Basis

Existing TS Section 3/4.5, "Core and Containment Cooling Systems," contains LCOs and SRs that provide assurance of adequate cooling capability for heat removal in the event of a LOCA or isolation from the normal reactor heat sink. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply in a defueled condition are being proposed for deletion.

10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," specifies that light-water nuclear power reactors must have ECCS designed to meet the cooling performance requirements following postulated LOCAs. 10 CFR 50.46(a)(1)(i) states "This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted."

<u>Core Spray and Low Pressure Coolant Injection</u>

TS 3/4.5.A specifies the operability requirements for the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) systems. This specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the reactor vessel. The CS and LPCI systems provide sufficient cooling to the core to dissipate the energy associated with the LOCA. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. Therefore, this specification will no longer be necessary and may be deleted.

<u>Containment Spray Cooling Capability</u>

TS 3/4.5.B specifies the operability requirements for the containment spray cooling capability and RHR Service Water system in order to ensure the ability to remove heat energy from the containment in the event of a LOCA. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. Therefore, this specification will no longer be necessary and may be deleted.

Residual Heat Removal Service Water System

TS 3/4.5.C specifies the operability requirements for the Residual Heat Removal (RHR) Service Water system in order to ensure the ability to remove heat energy from the containment in the event of a LOCA. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. Therefore, this specification will no longer be necessary and may be deleted.

Station Service Water and Alternate Cooling Tower Systems

TS 3/4.5.D specifies the operability requirements for the Station Service Water (SW) and Alternate Cooling Tower (ACS) systems during reactor operation. The SW system essential equipment cooling loops provide redundant heat sinks to dissipate residual heat in the reactor fuel after a shutdown or accident. The ACS provides the necessary heat sink for normal post-shutdown conditions in the event that the SW system becomes incapacitated. Once VY is in the permanently defueled condition, the normal post-shutdown and DBA conditions will no longer be possible. Therefore, this specification will no longer be necessary and may be deleted.

High Pressure Coolant Injection System

TS 3/4.5.E specifies the operability requirements for the High Pressure Coolant Injection (HPCI) system. HPCI is provided to adequately cool the core for all pipe breaks (i.e. LOCA) smaller than those for which the LPCI or CS subsystems can protect the core. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. Therefore, this specification will no longer be necessary and may be deleted.

Automatic Depressurization System

TS 3/4.5.F specifies the operability requirements for the Automatic Depressurization System (ADS). The ADS is designed to provide depressurization of the reactor coolant system during a small break LOCA if HPCI fails or is unable to maintain sufficient reactor water level. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. Therefore, this specification will no longer be necessary and may be deleted.

Reactor Core Isolation Cooling System (RCIC)

TS 3/4.5.G specifies the operability requirements for the RCIC system. RCIC is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the emergency core cooling systems. Once VY is in the permanently defueled condition, the conditions for which RCIC is required to operate will no longer be possible. Therefore, this specification will no longer be necessary and may be deleted.

Minimum Core and Containment Cooling System Availability

TS 3/4.5.H specifies the minimum core and containment cooling system availability. The core cooling and containment cooling subsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that the core and containment cooling function is maintained with any combination of allowed inoperable components. Once VY is in the permanently defueled condition, the conditions for which minimum system availability must be maintained will no longer be possible. Therefore, this specification will no longer be necessary and may be deleted. Note that TS 4.5.H was previously deleted.

Maintenance of Filled Discharge Pipe

TS 3/4.5.I specifies the requirements maintaining the discharge lines of the core spray subsystems, LPCI subsystems, HPCI and RCIC full of water to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start. Because none of these systems are proposed for inclusion in the PDTS, as described above, this specification will no longer be necessary in the permanently defueled condition and may be deleted.

Summary

The protection and monitoring functions of the Core and Containment Cooling Systems have been designed to ensure adequate heat removal capability following a LOCA. Once VY dockets 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 because a LOCA is not a postulated DBA in this condition. Additionally, the requirements of 10 CFR 50.46 will no longer be applicable. Based on the above, the proposed deletion of TS Section 3/4.5 in its entirety is acceptable.

TS SECTION 3/4.6, REACTOR COOLANT SYSTEM

Current TS 3/4.6	Proposed TS 3/4.6
A. Pressure and Temperature Limitations	A. Deleted
B. Coolant Chemistry	B. Deleted
C. Coolant Leakage	C. Deleted
D. Safety and Relief Valves	D. Deleted
E. Structural Integrity and Operability Testing	E. Deleted
F. Jet Pumps	F. Deleted
G. Single Loop Operation	G. Deleted
H. Recirculation System	H. Deleted
J. Thermal Hydraulic Stability	J. Deleted

Basis

Existing TS Section 3/4.6, “Reactor Coolant System,” contains LCOs and SRs that provide assurance of the integrity and safe operation of the RCS and the operation of the related safety devices. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply (or are no longer needed) in a defueled condition are being proposed for deletion. Note that TS 3/4.6.I has previously been deleted.

<u>Pressure and Temperature Limitations</u>

TS 3/4.6.A specifies maximum specified heatup and cooldown rates during reactor startup and shutdown to be consistent with the design assumptions and satisfy the stress limits for cyclic operation. These limitations are not required for a plant that will be in the permanently defueled condition since the Part 50 license will prohibit reactor operation and may be deleted. Accordingly, TS Figures 3.6.1, 3.6.2 and 3.6.3, which are referenced in TS 3.6.A and contain the reactor vessel pressure-temperature limitations may also be deleted.

<u>Coolant Chemistry</u>

TS 3/4.6.B specifies the radiochemistry limits for the reactor coolant in the RCS. Limits on the maximum allowable level of radioactivity in the coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 50.67. This specification may be deleted because it is not applicable to the safe storage and handling of spent fuel when the plant is in the permanently defueled condition. Additionally, the FHA, which will be the only credible DBA, will not result in a release of reactor coolant to the environment.

<u>Coolant Leakage</u>

TS 3/4.6.C specifies the maximum leakage rates of reactor coolant from the RCS. This specification may be deleted because it is not applicable the safe storage and handling of spent fuel when the plant is in the permanently defueled condition.

<u>Safety and Relief Valves</u>

TS 3/4.6.D specifies the operability requirements for the safety and relief valves. These valves provide overpressure protection to the reactor during operation. The Part 50 license will prohibit operation of the reactor once the certifications required by 10 CFR 50.82(a)(1) have been submitted. Therefore, this specification may be deleted.

<u>Structural Integrity and Operability Testing</u>

TS 3/4.6.E specifies the requirements for structural integrity and operability testing of safety-related components.

The inservice inspection and testing programs will no longer be required in the permanently defueled condition. 10 CFR 50.55a(f) and (g) are applicable only to operating nuclear power reactors. Once VY is certified to be permanently defueled, it will no longer be an operating power reactor.

As discussed in Regulatory Guide 1.184, “Decommissioning of Nuclear Power Reactors,” the maintenance rule, 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” requires monitoring the performance or condition of certain SSCs that could affect safety. Once the certifications for cessation of operation and for permanent fuel removal specified in 10 CFR 50.82(a)(1) have been submitted, the maintenance rule will be used to monitor the performance or condition of the SSCs associated with the storage, control, and maintenance of spent fuel in a safe condition. The monitoring will be sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions as

TS SECTION 3/4.7, STATION CONTAINMENT SYSTEMS

Current TS 3/4.7	Proposed TS 3/4.7
A. Primary Containment	A. Deleted
B. Standby Gas Treatment System	B. Deleted
C. Secondary Containment System	C. Deleted
D. Primary Containment Isolation Valves	D. Deleted
E. Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)	E. Deleted

Basis

The existing TS Section 3/4.7, "Station Containment Systems," contains LCOs and SRs that assure the integrity of the primary and secondary containment systems. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply in a defueled condition are being proposed for deletion.

Primary Containment

TS 3/4.7.A specifies the operability requirements for primary containment. The safety objective of the primary containment system in conjunction with core standby cooling systems is to provide the capability, in the event of the postulated LOCA, to limit the release of fission products to the plant environs so that off-site doses would be well below the values specified in 10 CFR 50.67. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. The FHA analysis does not rely on primary containment to mitigate the consequences of an FHA. Therefore, this specification will no longer be necessary and may be deleted.

Standby Gas Treatment System

TS 3/4.7.B specifies the operability requirements for the Standby Gas Treatment (SGT) system. This system is designed to filter and exhaust the Reactor Building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the Reactor Building to the environs.

With both trains of the SGT inoperable, the changes proposed by Reference 2 would prohibit movement of “recently irradiated” fuel and operations with the potential to drain the reactor vessel. Following approval of the changes proposed by Reference 2, after 13 days of decay following shutdown, the nuclear fuel will no longer be considered to be “recently irradiated.” The other condition that would be prohibited (operations with the potential to drain the reactor vessel) in the event that both trains of the SGT system are inoperable will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, these prohibited conditions will no longer be applicable.

The new FHA-based analysis provided in Reference 2 shows that after 13 days of fuel decay, secondary containment and the SGT system will not be required to mitigate the consequences of the FHA, which is the only DBA that will be possible in the permanently defueled condition. As a result, this specification will no longer be necessary and may be deleted.

Secondary Containment System

TS 3/4.7.C specifies the operability requirements for secondary containment. The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The Reactor Building provides secondary containment during reactor operation, when the drywell is sealed and in service; the Reactor Building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling.

In order for secondary containment integrity to be met, the secondary containment must function properly in conjunction with the operation of the SGT system to ensure that the required vacuum can be established and maintained. This means that the reactor building is intact with at least one door in each access opening closed, and all reactor building automatic ventilation system isolation valves are operable or the affected penetration flow path is isolated.

Following approval of the changes proposed by Reference 2, after 13 days of decay following shutdown, the nuclear fuel will no longer be considered to be “recently irradiated.” The other condition requiring that secondary containment integrity be met (operations with the potential to drain the reactor vessel) will not be applicable following permanent removal of the fuel from the reactor vessel. Therefore, the conditions requiring secondary containment integrity will no longer be applicable and secondary containment will not be required to mitigate the consequences of the FHA, which is the only DBA that will be possible in the permanently defueled condition. As a result, this specification will no longer be necessary and may be deleted.

Primary Containment Isolation Valves

TS 3/4.7.D specifies the operability requirements for the primary containment isolation valves. The primary containment design includes lines that penetrate the primary containment with different containment isolation valve configurations including double and single valve isolation. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA. Once VY is in the permanently defueled condition, the only postulated DBA that will be able to occur is the FHA. The FHA analysis does not rely on primary containment to mitigate the consequences of an FHA. Therefore, this specification will no longer be necessary and may be deleted.

Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

3/4.7.E specifies the operability requirements for the RBAVSIVs when secondary containment integrity is required. The function of the RBAVSIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated DBAs. The operability requirements for RBAVSIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. The RBAVSIVs must be operable (or the penetration flow path isolated) to ensure secondary containment integrity and to limit the potential release of fission products to the environment.

Following approval of the changes proposed by Reference 2, after 13 days of decay following shutdown, the nuclear fuel will no longer be considered to be “recently irradiated” and secondary containment integrity will no longer be required to mitigate the consequences of the FHA, which is the only DBA that will be possible in the permanently defueled condition. As a result, this specification will no longer be necessary and may be deleted.

Summary

The analyzed DBA that remains applicable to VY in the permanently shut down and defueled condition is a FHA. Section 5.0(a) of the new FHA-based analysis in Reference 2 shows that, following 13 days of fuel decay time after reactor shutdown, the dose consequences are acceptable without relying on SSCs remaining functional for accident mitigation during and following the event. The exception to this is the continued function of the passive spent fuel pool structure. Proposed changes to TS 3.7.B.4 and TS 3.7.C described in Reference 2 are under NRC review. Following approval of the changes proposed in Reference 2 and following 13 days of fuel decay, the specifications associated with secondary containment will no longer be necessary.

Once VY dockets 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, the specifications associated with primary containment will not be needed once VY is in the permanently defueled condition and is no longer applicable.

Based on the above, the proposed deletion of TS Section 3/4.7 in its entirety is acceptable.

TS SECTION 3/4.8, RADIOACTIVE EFFLUENTS

*(Note: This section appears renumbered as new TS Section 3/4.1 in the markup and retyped TS pages. Proposed TS 3/4.8.D is renumbered to TS 3/4.1.A)

Current TS 3/4.8	Proposed TS 3/4.8*
A. Deleted	
B. Deleted	
C. Deleted	
D. Liquid Holdup Tanks	D. No Change*
E. Deleted	
F. Deleted	
G. Deleted	
H. Deleted	
I. Deleted	
J. Explosive Gas Mixture	J. Deleted
K. Steam Jet Air Ejector (SJAE)	K. Deleted

L. Deleted M. Deleted N. Deleted	

Basis

The existing TS Section 3/4.8, "Radioactive Effluents," contains LCOs and SRs that assure that radioactive effluents are kept “as low as is reasonably achievable” in accordance with 10 CFR 50, Appendix I and, in any event, are within the dose limits for Members of the Public specified in 10 CFR 20 and applies to the release of all radioactive effluents from the plant. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply in a defueled condition are being proposed for deletion. All previously deleted section placeholders will be administratively deleted. This change also corrects a typographical error (as shown in the markup in Attachment 2) in the title of current TS 3.8.

<u>Liquid Holdup Tanks</u>

TS 3/4.8.D specifies the allowed quantities of radioactive material in the liquid holdup tanks. The tanks listed in this specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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

This specification will continue to be used in the permanently defueled condition to ensure compliance with 10 CFR 20.1001-20.2402, but will be renumbered to TS 3/4.1.A. The table of contents is also revised to reflect the numbering changes.

<u>Explosive Gas Mixture</u>

TS 3/4.8.J specifies the limits and monitoring requirements for hydrogen concentration in the off-gas downstream of the operating recombiner in the Advanced Off-Gas System. The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Once the reactor is in the permanently defueled condition, hydrogen will no longer be produced in the RCS by reactor operation. Therefore, this specification will no longer be necessary.

<u>Steam Jet Air Ejector (SJAE)</u>

TS 3/4.8.K specifies limits on the gross radioactivity release rate of gases from the SJAE to provide reasonable assurance that the total effective dose equivalent to an individual at the exclusion area boundary will not exceed the limits of 10 CFR 50.67 in the event this effluent is inadvertently discharged directly to the environment without treatment. SJAEs are used to remove gases from the main condenser during normal reactor operation. Once the reactor is permanently defueled, the SJAEs will no longer be required. Therefore, this specification will no longer be necessary.

Summary

Once VY dockets 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, the Specifications for explosive gas mixture and SJAEs will not be needed and will no longer be applicable once VY is in the permanently defueled condition. Based on the above, the proposed deletion of TS 3/4.8.J and K is acceptable.

TS SECTION 3/4.9 was previously deleted

TS SECTION 3/4.10, AUXILIARY ELECTRICAL POWER SYSTEMS

Current TS 3/4.10	Proposed TS 3/4.10
A. Normal Operation	A. Deleted
B. Operation With Inoperable Components	B. Deleted
C. Diesel Fuel	C. Deleted

Basis

The existing TS Section 3/4.10, "Auxiliary Electrical Power Systems," contains LCOs and SRs that assure an adequate supply of electrical power for operation of those systems required for reactor safety. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply (or are no longer needed) in a defueled condition are being proposed for deletion.

Normal Operation

TS 3/4.10.A specifies requirements to ensure that the offsite power sources, the onsite standby power sources (EDGs, station batteries), emergency buses and RPS power protection provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems during and following any DBA so that the fuel, RCS, and containment design limits are not exceeded.

This specification assures that at least two off-site and two on-site power sources, and both 480 V Uninterruptible Power Systems will be available before the reactor is made critical. In addition to assuring power source availability, all of the associated switchgear must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.

The design basis accidents and transients analyzed in UFSAR Section 14 will no longer be applicable in the permanently defueled condition, with the exception of the FHA. The new FHA-based analysis shows that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the event (following 13 days of irradiated fuel decay time after reactor shutdown and compliance with the SFP water level and temperature requirements of TS 3/4.12).

Because the FHA analysis does not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of irradiated fuel assemblies for mitigation of a potential FHA. Therefore, during movement of irradiated fuel assemblies, there are no active systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA that is credible with the unit permanently defueled. As such, the requirement for AC sources is being deleted because there are no design basis events that rely on AC sources for mitigation.

Because the FHA analysis does not rely on DC sources for accident mitigation (dose consequences are acceptable without relying on any SSCs to remain functional during and following the event), DC sources are therefore not required for accident mitigation. Consequently, DC sources are not needed during movement of irradiated fuel assemblies for mitigation of a potential FHA. Thus, the requirement for DC sources is being deleted.

The preceding TS are related to assuring an adequate supply of electrical power for operation of those systems required for reactor safety. However, 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel. Therefore, the TS listed in the previous paragraphs, which only address these specific plant systems, will no longer be applicable once the facility is in the permanently defueled condition. Based on the above, the proposed deletion of TS related to these electrical systems is acceptable.

Operation With Inoperable Components

TS 3/4.10.B specifies requirements that ensure adequate power is available to operate the emergency safeguards equipment from the immediate access source or for minimum engineered safety features from either of the EDGs. This specification is only applicable when the reactor is in the RUN or STARTUP modes with the reactor not in the Cold Condition. Once VY is in the permanently defueled condition, this specification will no longer be applicable because the VY Part 50 license will prohibit reactor operation.

Because of the continued need for normal electrical power to supply equipment needed for cooling of irradiated fuel stored in the SFP, and as defense in depth, the Technical Requirements Manual (TRM) (which is subject to the requirements of 10 CFR 50.59) will retain a requirement for a single functional circuit between the offsite transmission network and the (non-Class 1E) onsite AC electrical power distribution subsystems. Although there is no need for emergency power in order to fulfill the SFP cooling function, a backup AC power source will be maintained for defense in depth. The requirement for a single circuit between the offsite transmission network and the onsite AC distribution subsystems will be applicable whenever any irradiated fuel assembly is stored in the spent fuel pool and also applicable during movement of irradiated fuel assemblies.

Battery Room Ventilation and Sampling (TS 3/4.10.B.2.a)

The Battery Room is ventilated to prevent accumulation of hydrogen gas. With a complete loss of the ventilation system, the accumulation of hydrogen would not exceed 4 percent concentration in 2 1/2 days. Therefore, on loss of Battery Room ventilation, the use of portable ventilation equipment and daily sampling provide assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

Hydrogen generation can still occur inside the batteries even if the plant is permanently defueled. Although reliance on DC batteries is no longer required for accident mitigation as described above, the potential hydrogen fire and explosion hazard still exists when the reactor is permanently defueled. Therefore, this specification and surveillance will be moved to the TRM to meet the requirements of 10 CFR 50.48(f).

Diesel Fuel

TS 3/4.10.C specifies parameters for stored diesel fuel oil. Stored diesel fuel oil is required to have sufficient supply for 7 days of rated load operation for a single EDG. It is also required to meet specific standards for quality. These requirements support the availability of the EDGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of off-site power. This specification will no longer be necessary once the reactor is permanently defueled because there will no postulated AOOs or DBAs which can occur at VY which will require the EDGs for mitigation.

With the reactor permanently defueled, irradiated fuel is stored either on the ISFSI or in the SFP. The ISFSI is a passive system that does not rely on electrical power for heat transfer. Since there is a large capacity for heat absorption in the SFP, active system components are not redundant. Alternate cooling capability can be made available during anticipated malfunctions or failures without reliance on EDGs.

Summary

As discussed in the justification for deleting TS 3.10.A.1 above, the requirement for EDGs is being deleted from the TS because there will no longer be any DBAs or transients analyzed in USFAR Section 14 that rely on the EDGs for mitigation once VY dockets the certifications required by 10 CFR 50.82(a)(1). Since TS 3/4.10.C exists solely to support the EDG requirements of TS 3/4.10.A.1 and TS 3/4.10.B.1, the elimination of the need for EDGs also obviates the need for their support systems. As such, TS 3/4.10.C may be deleted. Based on the above, the proposed deletion of TS 3/4.10.C for fuel oil parameters is acceptable.

Because of the continued need for electrical power to supply equipment needed for cooling of irradiated fuel stored in the spent fuel pool and as defense in depth, functionality requirements for the offsite AC power source are being added to the TRM that will be applicable whenever any irradiated fuel assembly is stored in the SFP and also during movement of irradiated fuel assemblies. TRM requirements are also being added to monitor battery room ventilation.

TS SECTION 3/4.11, REACTOR FUEL ASSEMBLIES

Current TS 3/4.11	Proposed TS 3/4.11
A. Average Planar Linear Heat Generation Rate (APLHGR)	A. Deleted
B. Linear Heat Generation Rate (LHGR)	B. Deleted

C. Minimum Critical Power Ratio (MCPR)	C. Deleted

Basis

The existing TS 3/4.11, "Reactor Fuel Assemblies," contains LCOs and SRs associated with the parameters that monitor condition of the nuclear fuel during reactor operations to assure adequate fuel performance and are being proposed for deletion in their entirety. These requirements will no longer apply once the reactor is certified to be permanently defueled because the VY Part 50 license will no longer allow placement of the fuel in the reactor vessel.

Average Planar Linear Heat Generation Rate (APLHGR)

TS 3/4.11.A specifies APLHGR limits to ensure that the peak cladding temperature during a design basis LOCA does not exceed 2200ºF. Since a LOCA will not be a credible accident in the permanently defueled condition and the Part 50 license will prohibit reactor operation, this specification will no longer be required.

Linear Heat Generation Rate (LHGR)

TS 3/4.11.B specifies requirements for the linear heat generation rate of a fuel rod at a given nodal plane in a bundle. These limits are bundle type dependent and monitored to assure all mechanical design requirements are met. Since the Part 50 license will prohibit reactor operation in the permanently defueled condition, this specification will no longer be required.

Minimum Critical Power Ratio (MCPR)

TS 3/4.11.C specifies the MCPR operating limit, which is a cycle-dependent parameter that can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provide reasonable assurance against exceeding the Fuel Cladding Integrity Safety Limit for potential abnormal occurrences. Since the Part 50 license will prohibit reactor operation in the permanently defueled condition, this specification will no longer be required.

Summary

The protection and monitoring functions of TS 3/4.11, Reactor Fuel Assemblies, have been designed to ensure safe operation of the fuel in the reactor. Once VY dockets 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 and will no longer be applicable once VY is in the permanently defueled condition. Based on the above, the proposed deletion of TS Section 3/4.11 in its entirety is acceptable.

TS SECTION 3/4.12, REFUELING AND SPENT FUEL HANDLING

*(Note: This section appears renumbered as new TS Section 3/4.2 in the markup and retyped TS pages. Proposed TS 3/4.12.C is renumbered to TS 3/4.2.A and proposed TS 3/4.12.H is renumbered to TS 3/4.2.B.)

Current TS 3/4.12, REFUELING AND SPENT FUEL HANDLING	Proposed TS 3/4.12*, SPENT FUEL STORAGE
3.12 Applicability: Applies to fuel handling, core reactivity limitations, and spent fuel handling.	3.12* Applicability: Applies to storage of spent fuel.
3.12 Objective: To assure core reactivity is within capability of the control rods, to prevent criticality during refueling, and to assure safe handling of spent fuel casks.	3.12* Objective: To assure safe storage of spent fuel.
4.12 Applicability: Applies to the periodic testing of those interlocks and instruments used during refueling.	4.12* Applicability: Applies to the parameters which monitor the storage of spent fuel.
4.12 Objective: To verify the operability of instrumentation and interlocks used in refueling.	4.12* Objective: To verify that spent fuel is being stored safely.
3/4.12.A Refueling Interlocks	3/4.12.A Deleted
3/4.12.B Core Monitoring	3/4.12.B Deleted
3/4.12.C Fuel Storage Pool Water Level	3/4.12.C* No Change
3/4.12.D Control Rod and Control Rod Drive Maintenance	3/4.12.D Deleted
3/4.12.E Extended Core Maintenance	3/4.12.E Deleted
3/4.12.F Fuel Movement	3/4.12.F Deleted
3/4.12.G Deleted	3/4.12.G Deleted
3.12.H Spent Fuel Pool Water Temperature	3.12.H* No Change
4.12.H Spent Fuel Pool Water Temperature	4.12.H* Spent Fuel Pool Water Temperature
Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be recorded daily. If the pool water temperature reaches 150°F, all refueling operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.	Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be recorded daily. If the pool water temperature reaches 150°F, all operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.
Basis	

The existing TS Section 3/4.12, "Refueling and Spent Fuel Handling," contains LCOs and SRs that provide for appropriate functional capability of parameters and equipment within containment that are required for mitigation of design basis accidents during refueling operations (moving irradiated fuel to or from the reactor core). Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs (and associated SRs) that do not apply in a defueled condition are being proposed for deletion. The title of this section is proposed to be changed to “Spent Fuel Storage” to reflect that refueling of the reactor will no longer be possible. The applicability and objectives are also proposed to be changed to reflect the focus on safe storage of spent fuel. This specification is being renumbered to TS 3/4.2. The table of contents is also revised to reflect the numbering changes.

Refueling Interlocks

TS 3/4.12.A specifies requirements for refueling interlocks and restriction of certain refueling procedures such that there is assurance that inadvertent criticality does not occur during refueling. Refueling will not be permitted by the Part 50 license following certification of permanent removal of the fuel. Therefore, this specification will no longer be required.

Core Monitoring

TS 3/4.12.B specifies requirements for monitoring the core during periods of station shutdown and to guide the operator during refueling operations and station startup. When all of the fuel has been permanently removed from the reactor, core monitoring will no longer be required.

Fuel Storage Pool Water Level

TS 3/4.12.C specifies requirements to assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. This minimum water level of 36 feet is established because it would be a significant change from the normal level, well above a level to assure adequate cooling (just above active fuel). Since the reactor core will be offloaded to the SFP following shutdown, this specification will remain with no changes except for being renumbered to TS 3/4.2.A.

Control Rod and Control Rod Drive Maintenance

TS 3/4.12.D specifies requirements for performing control rod/control rod drive maintenance and maintaining adequate shutdown margin during such activities to provide assurance that inadvertent criticality does not occur during such maintenance. With all fuel permanently removed from the reactor, this specification will no longer be required.

Extended Core Maintenance

TS 3/4.12.E specifies requirements for unloading of a portion of the reactor core for such purposes as in-service inspection requirements, examination of the core support plate, control rod, control rod drive maintenance, etc. This specification provides assurance that inadvertent criticality does not occur during such operation. With all fuel permanently removed from the reactor, this specification will no longer be required.

Fuel Movement

TS 3/4.12.G specifies requirements to assure that the reactor core has been shut down for at least 24 hours following power operation and prior to fuel handling or movement. The analysis (Reference 2) based on the postulated FHA assumed that the reactor had been shut down for 24 hours for fission product decay prior to any fuel handling which could result in dropping of a fuel assembly. With all fuel permanently removed from the reactor, this specification will no longer be required.

TS SECTION 3/4.13 was previously deleted

TS SECTION 5.0, DESIGN FEATURES

*(Note: Proposed TS Section 5.5 appears renumbered as new TS Section 5.2 in the markup and retyped TS pages. Proposed TS 5.5.B-E is renumbered to TS 5.2.A-D.)

Current TS 5.0	Proposed TS 5.0
5.1 Site	5.1 No Change
5.2 Reactor	5.2 Deleted
5.3 Reactor Vessel	5.3 Deleted
5.4 Containment	5.4 Deleted
5.5 Spent and New Fuel Storage	5.5 *Spent Fuel Storage
A. The new fuel storage facility shall be such that the effective multiplication factor (k_{eff}) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.	A. Deleted
B. The k_{eff} of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.	B. *No Change
C. Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.	C. *No Change
D. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3353.	D. *No Change
E. The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool or the new fuel storage facility shall be less than or equal to 1.31 at 20°C.	E. *The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool shall be less than or equal to 1.31 at 20°C.

Basis

The existing TS Section 5.0, “Design Features,” contains descriptions and requirements for those features of the facility such as materials of construction and geometric arrangements which, if altered or modified, could have a significant effect on safety and are not covered in the previous sections of the TS. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the design features that do not apply in a defueled condition are being proposed for deletion.

TS 5.1, Site Location, provides a description regarding the location of VY. This TS section is being retained in the permanently defueled TS with no changes to the TS.

TS 5.2, Reactor, provides a description and requirements regarding the reactor core fuel assemblies and control rod assemblies. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, this TS section will not apply in a defueled condition and is being proposed for deletion.

TS 5.3, Reactor Vessel, provides a reference to the FSAR description of the reactor vessel and applicable design codes. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, this TS section will not apply in a defueled condition and is being proposed for deletion.

TS 5.4, Containment, provides references to principal design parameters and applicable design codes for the primary containment, the secondary containment and applicable codes, and design standards for penetrations to the primary containment and piping passing through such penetrations. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, this TS section will not apply in a defueled condition and is being proposed for deletion.

TS 5.5.A describes criticality restrictions for the new fuel storage facility. The new fuel storage facility will not be used to store spent fuel and new fuel is no longer stored onsite. License Condition 2.B is being revised to no longer allow receipt of new fuel. Therefore, this TS section will not apply in the permanently defueled condition and is being proposed for deletion. Accordingly, the title of TS Section 5.5 is proposed to be revised to reflect storage of spent fuel is all that is permitted at VY once it is in the permanently defueled condition.

TS 5.5.B, C, D and E provides a description and requirements regarding prevention of criticality of spent fuel, prevention of spent fuel damage when moving spent fuel racks, and SFP capacity limitations. These TS sections are being retained as-is in the permanently defueled TS, with the exception of deleting the new fuel storage facility criticality limitation in TS 5.5.D. These specifications are being renumbered to TS 5.2.A-D.

<u>Summary</u>

TS 5.1 and TS 5.5 (with the exception of TS 5.5.A) will remain applicable with the reactor permanently defueled. As such, these TS sections (with the exception of TS 5.5.A) are being retained to reflect a permanently defueled condition. Because License Condition 2.B is being revised to no longer allow receipt of new fuel, TS 5.5.A is being deleted and TS 5.5.E is revised to delete reference to the new fuel storage facility. These changes will have no impact on continued safe storage of spent fuel.

The protection and monitoring functions of TS 5.0, Design Features have been designed to ensure safe handling of the fuel in the reactor and in the SFP during refueling and in the new fuel storage facility. Once VY dockets 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, other than the site description and the SFP storage restrictions, will not be needed once VY is in the permanently defueled condition. Based on the above, the proposed deletion of the noted TS in Section 5.0 is acceptable.

TS SECTION 6.0, ADMINISTRATIVE CONTROLS

Administrative controls were proposed for revision or deletion in a letter dated October 31, 2013 (Reference 3). An additional change is proposed to the TS table of contents to remove TS sections 6.1 and higher from the table of contents so that only TS 6.0 is shown. This is an editorial and administrative change.

3. REGULATORY EVALUATION

3.1 APPLICABLE REGULATORY REQUIREMENTS

10 CFR 50.82, Termination of License

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 permanently cease power operations 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. VY recognizes that approval of these proposed changes is contingent upon the docketing 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."

Other applicable regulatory requirements are:

10 CFR 50.36, Technical Specifications

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of Technical Specifications (TS). In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." (Statement of

Consideration, "Technical Specification for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968)) Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

These criteria, which were subsequently codified in changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953), also pertain to the Technical Specification requirements for safe storage of spent fuel. A general discussion of these considerations is provided below.

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

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some DBAs which continue to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The applicable DBA for VY in the permanently defueled condition, the FHA, is discussed within this proposed amendment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The intent of this criterion is to capture into TS only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that will continue to apply to VY, there are still DBAs that will continue to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant. The scope of DBAs that will be applicable to VY is discussed in more detail within this proposed amendment.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. All of the accident sequences that previously dominated risk at VY will no longer be applicable once the reactor is in the permanently shutdown and defueled condition.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they "...are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The particular administrative controls to be included in the TS, therefore, are the provisions that the Commission deems essential for the safe operation of the facility that are not already covered by other regulations. Accordingly, the NRC staff determined that administrative control requirements that are not specifically required under Section 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or an event giving rise to an immediate threat to the public health and safety, may be relocated to more appropriate documents (e.g., Quality Assurance Program, Security Plan, or Emergency Plan), which are subject to regulatory controls. Similarly, while the required content of TS administrative controls is specified in 10 CFR 50.36(c)(5), particular details may be relocated to licensee-controlled documents, where other regulations provide adequate regulatory control.

10 CFR 50.36(c)(6), "Decommissioning," applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1). For such facilities, TS involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

This proposed amendment deletes the portions of the previous VY TS that are no longer applicable to a permanently defueled facility while modifying the remaining portions to correspond to the permanently shutdown condition.

<u>10 CFR 50.48(f), Fire Protection during Decommissioning</u>

10 CFR 50.48(f) states in part that licensees that have submitted the certifications required under § 50.82(a)(1) shall maintain a fire protection program to address the potential for fires that could cause the release or spread of radioactive materials (i.e., that could result in a radiological hazard).

(1) The objectives of the fire protection program are to--
- (i) Reasonably prevent these fires from occurring;
- (ii) Rapidly detect, control, and extinguish those fires that do occur and that could result in a radiological hazard; and
- (iii) Ensure that the risk of fire-induced radiological hazards to the public environment and plant personnel is minimized.

(2) The licensee shall assess the fire protection program on a regular basis. The licensee shall revise the plan as appropriate throughout the various stages of facility decommissioning.

(3) The licensee may make changes to the fire protection program without NRC approval if these changes do not reduce the effectiveness of fire protection for facilities, systems, and equipment that could result in a radiological hazard, taking into account the decommissioning plant conditions and activities.

10 CFR 50.51, Continuation of License

10 CFR 50.51(b) states "Each license for a facility that has permanently ceased operations, continues in effect beyond the expiration date to authorize ownership and possession of the production or utilization facility, until the Commission notifies the licensee in writing that the license is terminated. During such period of continued effectiveness the licensee shall:

(1) Take actions necessary to decommission and decontaminate the facility and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition, and
(2) Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the provisions of the specific 10 CFR part 50 license for the facility."

10 CFR 50.54(hh), Loss Due to Aircraft Threats and Explosion or Fire

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).

10 CFR 50.82, Termination of License

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, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants

The GDC in place today became effective after the VY construction permit was issued. A September 18, 1992 memorandum (ML003763736) to the NRC Executive Director of Operations from the Secretary of the NRC summarized the results of a Commissioners vote in which the Commissioners instructed the NRC staff not to apply the GDC to plants with construction permits issued prior to May 21, 1971. VY's construction permit was issued on December 11, 1967.

VY's design and licensing basis for fuel storage and handling and radiological controls is detailed in the UFSAR and other plant-specific licensing basis documents. UFSAR Appendix F provides a comparison of VY with the proposed GDC published by the Atomic Energy Commission (AEC) for public comment in the Federal Register dated July 11, 1967.

10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

10 CFR 50.46(a)(1)(i) states “This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted.”

10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

10 CFR 50.62(a) states "The requirements of this section apply to all commercial light-water-cooled nuclear power plants, other than nuclear power reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted."

Design Basis Accidents (DBAs)

Section 14 of the VY UFSAR describes the safety analysis aspects of the plant that were evaluated to demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed the guidelines of 10 CFR 50.67. Two basic groups of events pertinent to safety, abnormal operational transients, and accidents, will be investigated separately. The safety design limits require that damage to the fuel be limited and that no nuclear system process barrier damage results from any abnormal operational transient. Thus, analysis of this group of events evaluates the features that protect the first two radioactive material barriers. Analysis of the events in the second group, accidents, evaluates situations that require functioning of the engineered safeguards, including containment.

Safety analyses are analyses performed to satisfy regulatory requirements. The safety analyses are integral to the plant's design and licensing basis. The safety analyses demonstrate the integrity of the fission product barriers, the capability to shutdown the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents and transients. SSCs that perform design basis functions are credited in the safety analyses for the purpose of mitigating the transient or accident.

Chapter 14 of the VY UFSAR describes the DBA scenarios that are applicable during plant operations. However, 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, as specified in 10 CFR 50.82(a)(2). Therefore, most of the accident scenarios postulated in UFSAR Section 14 will no longer be applicable once VY is in the permanently defueled condition.

3.2 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Pursuant to 10 CFR 50.92, 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 10 CFR 50.92(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 amendment would revise the Vermont Yankee Nuclear Power Station (VY) Renewed Facility Operating License (OL) and associated Technical Specifications (TS) to Permanently Defueled Technical Specifications (PDTS) to reflect the pending permanent cessation of reactor operation.

On August 27, 2013, ENO announced that VY would permanently cease power operations 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 pursuant to 10 CFR 50.82(a)(1)(i). 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 existing VY TS contain limiting conditions for operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Because the VY Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel, the LCOs and associated surveillance requirements (SRs) that will not apply in the permanently defueled condition are being proposed for deletion. The remaining portions of the TS are being proposed for revision, into the PDTS, to provide an acceptable level of safety derived from the reduced scope of postulated design basis accidents associated with a defueled plant, as described in the VY safety analyses.

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. This change is consistent with the criteria set forth in 10 CFR 50.36 for the contents of TS.

   Section 14 of the VY Updated Final Safety Analysis Report (UFSAR) describes the design basis accident (DBA) and transient scenarios applicable to VY during power operations. Once the reactor is in a permanently defueled condition, the spent fuel pool and its cooling systems will be dedicated only to spent fuel storage. In this condition, the spectrum of credible accidents will be much smaller than for an operational plant. Once the certifications are docketed by VY in accordance with 10 CFR 50.82(a)(1), and the consequent removal of authorization to operate the reactor or to place or retain fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2), the majority of the accident scenarios previously postulated in the UFSAR will no longer be possible and will be removed from the UFSAR under the provisions of 10 CFR 50.59.

   The deletion of TS definitions and rules of usage and application, that will not be applicable in a defueled condition, has no impact on facility SSCs or the methods of operation of such SSCs. The deletion of design features and safety limits not applicable to the permanently shutdown and defueled status of VY has no impact on the remaining applicable DBA, the Fuel Handling Accident (FHA). The removal of LCOs or SRs that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents do not affect the applicable DBAs previously evaluated since these DBAs are no longer applicable in the defueled mode. The safety functions involving core reactivity control, reactor heat removal, reactor coolant system inventory control, and containment integrity are no longer applicable at VY as a permanently defueled plant. The analyzed accidents involving damage to the reactor coolant system, main steam lines, reactor core, and the subsequent release of radioactive material will no longer be possible at VY.

After VY permanently ceases operation, the future generation of fission products will cease and the remaining source term will decay. The radioactive decay of the irradiated fuel following shutdown of the reactor will have reduced the consequences of the FHA below those previously analyzed.

The spent fuel pool (SFP) water level, temperature and storage TSs are retained to preserve the current requirements for safe storage of irradiated fuel. SFP cooling and makeup related equipment and support equipment (e.g., electrical power systems) are not required to be continuously available since there will be sufficient time to effect repairs, establish alternate sources of makeup flow, or establish alternate sources of cooling in the event of a loss of cooling and makeup flow to the SFP.

The TS for outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system are retained to preserve the current requirements for safe storage of radioactive liquids. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an unrestricted area.

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 will no longer be 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 removal of TS that are related only to the operation of the nuclear reactor or only to the prevention, diagnosis, or mitigation of reactor-related transients or accidents, 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 be 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 FHA at VY. The proposed OL and TS will continue to require proper control and monitoring of safety significant parameters and activities.

The proposed restriction on the SFP level is fulfilled by normal operating conditions and preserves initial conditions assumed in the analyses of the postulated DBA. The SFP water level, temperature, and storage TSs are retained to preserve the current requirements for safe storage of irradiated fuel.

The TS for outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system are retained to preserve the current requirements for safe storage of radioactive liquids.

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 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 design basis accidents involving the reactor will no longer be 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 license and TS currently in effect for Millstone Nuclear Power Station (DPR-21), which was last substantively revised on March 31, 2001 (Reference 7). The Millstone license amendment that was issued to reflect the permanently shutdown status of the plant on November 9, 1999 (Reference 8), contains license conditions and TS similar to those being proposed herein.

The proposed changes are also consistent with the license and accompanying TS issued to Zion Nuclear Power Station on December 30, 1999 (Reference 9) 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 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section 3.2 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 that may be released offsite.

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

The proposed amendment does not involve any physical alterations to the plant configuration that could lead to 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 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of 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 (ADAMS Accession No. ML13273A204)

2. Letter, Entergy Nuclear Operations, Inc. to USNRC, “Technical Specifications Proposed Change No. 306, Eliminate Certain ESF Requirements during Movement of Irradiated Fuel,” BVY 13-097, dated November 14, 2013 (ADAMS Accession No. ML13323A518)

3. Letter, Entergy Nuclear Operations, Inc. to USNRC, “Technical Specifications Proposed Change No. 307, Revision to Mitigation Strategy License Condition and Technical Specification Administrative Controls for Permanently Defueled Condition,” BVY 13-096, dated October 31, 2013 (ADAMS Accession No. ML13316A004)

4. Letter, Entergy Nuclear Operations, Inc. to USNRC, “Request for Approval of Certified Fuel Handler Training Program,” BVY 13-095, dated October 31, 2013 (ADAMS Accession No. ML13325B015)

5. Letter, Entergy Nuclear Operations, Inc. to USNRC, “Request for Rescission of Security Orders,” BVY 13-105, dated December 18, 2013 (ADAMS Accession No. ML13365A007)

6. Letter, Entergy Nuclear Operations, Inc., to NRC, "Cyber Security Plan Implementation Schedule Proposed Change No. 308," BVY 13-109, dated December 19, 2013 (ADAMS Accession No. ML13358A338)

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

8. 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)

9. 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 309

Markup of the Current Operating License and Technical Specification Pages

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

(Vermont Yankee Nuclear Power Station)

Docket No. 50-271

Renewed Facility Operating License

Renewed Operating License No. DPR-28

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

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a. ~~Construction of the Vermont Yankee Nuclear Power Station (the facility) has been substantially completed in conformity with the application, as amended, the Provisional Construction Permit No. CPPR 36, the provisions of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in Title 10 of the *Code of Federal Regulations* (CFR) Chapter 1,; and~~

be prohibited from operating the reactor

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

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

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

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

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

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

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

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

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

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

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

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

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

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

E. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

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A. ~~Maximum Power Level~~

~~Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1912 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.~~

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

C. Reports

Entergy Nuclear Operations, Inc. shall make reports in accordance with the requirements of the Technical Specifications.

D. This paragraph deleted by Amendment No. 226.

E. Environmental Conditions

Pursuant to the Initial Decision of the presiding Atomic Safety and Licensing Board issued February 27, 1973, the following conditions for the protection of the environment are incorporated herein:

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

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

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

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

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

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

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

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

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

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

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

    (b) Total curie alpha activity discharged.

    (c) Total curies of tritium discharged.

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

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

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

(g) Average concentration at discharge outfall.

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

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

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

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

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

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

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

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

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

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

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F. ~~Entergy Nuclear Operations, Inc. shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final~~

~~Safety Analysis Report for the facility and as approved in the SER dated January 13, 1978, and supplemental SERs, dated 9/12/79, 2/20/80, 4/15/80, 7/3/80, 10/24/80, 11/10/81, 1/13/83, 7/24/84, 3/25/86, 12/1/86, 12/8/89, 11/29/90, 8/30/95, 3/23/97, 6/9/97, 8/12/97, 3/6/98, 3/31/98, 9/2/98, and 2/24/99, subject to the following provisions:~~

~~Entergy Nuclear Operations, Inc. may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.~~

G. Security Plan

Changes to License Condition 3.G, Security Plan, are under NRC review. Reference: Letter, Entergy Nuclear Operations, Inc., to NRC, "Cyber Security Plan Implementation Schedule Proposed Change No. 308," BVY 13-109, dated December 19, 2013

Entergy Nuclear Operations, Inc. shall fully implement
all provisions of the Commission-approved physical se
qualification, and safeguards contingency plans includ
pursuant to provisions of the Miscellaneous Amendme
Requirements revisions to 10 CFR 73.55 (51 FR 2781
authority of 10 CFR 50.90 and 10 CFR 50.54(p). The
which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Vermont Yankee Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 0," submitted by letter dated October 18, 2004, as supplemented by letter dated May 16, 2006.

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

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

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

J. License Transfer Conditions

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

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

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

a. Decommissioning Trust

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

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

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

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

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

b. Surety

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

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

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

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

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

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

K. ~~Minimum Critical Power Ratio~~ Deleted

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

L. ~~Transient Testing~~ Deleted

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

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

M. ~~Potential Adverse Flow Effects~~ Deleted

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

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

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

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

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

~~d. In addition to evaluating the MSL strain gage data, Entergy Nuclear Operations, Inc. shall monitor reactor pressure vessel water level instrumentation or MSL piping accelerometers on an hourly basis during power ascension above OLTP. If resonance frequencies are identified as increasing above nominal levels in proportion to strain~~

~~gage instrumentation data, Entergy Nuclear Operations, Inc. shall stop power ascension, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.~~

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

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

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

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

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

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

~~e. Entergy Nuclear Operations, Inc. shall revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of~~

~~steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with General Electric Services Information Letter 644, Revision 1; and to identify the NRC Project Manager for the facility as the point of contact for providing SDMP information during power ascension.~~

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

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

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

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

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

~~b. Level 1 performance criteria; and~~

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

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

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

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

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

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

Deleted

N. ~~Mitigation Strategy License Condition~~

This change is under NRC review. Reference: Letter, Entergy Nuclear Operations, Inc., to NRC, "Technical Specifications Proposed Change No. 307 Revision to Mitigation Strategy License Condition and Technical Specification Administrative Controls for Permanently Defueled Condition," BVY 13-096, dated October 31, 2013 (ML13316A004)

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

~~(a) Fire fighting response strategy with the followir~~
~~1. Pre-defined coordinated fire response stra~~
~~2. Assessment of mutual aid fire fighting ass~~
~~3. Designated staging areas for equipment a~~
~~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~~

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.

Deleted

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.~~

S. ~~Steam Dryer License Condition~~ Deleted

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

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

~~2.~~ ~~If the steam dryer inspection during RFO 31 or any subsequent steam dryer inspection reveals new cracking due to either fatigue or IGSCC that requires repair, or any growth of existing cracking that requires repair, the licensee shall inspect the steam dryer during each refueling outage.~~

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

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

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
Eric J. Leeds

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosures:
Appendix A - Technical Specifications

Date of Issuance: March 21, 2011

APPENDIX A

TO

OPERATING LICENSE DPR-28

TECHNICAL SPECIFICATIONS

AND BASES

FOR

VERMONT YANKEE NUCLEAR POWER STATION

VERNON, VERMONT

ENTERGY NUCLEAR OPERATIONS, INC.

AND

ENTERGY NUCLEAR VERMONT YANKEE, LLC

DOCKET NO. 50-271

~~Reissued by~~
~~Change Nos. 13, 15, and 17~~
~~Dated 1/17/74, 1/28/74, and 4/10/74~~

Amendment No. ~~208~~

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

1.0 DEFINITIONS

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

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

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

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

B

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

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

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

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

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

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

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

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

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

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

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

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

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

~~1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment, which are not required to be open during accident conditions, are closed. Such valves may be opened intermittently under administrative controls.~~

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

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

~~4. All blind flanges and manways are closed.~~

~~O. Protective Instrumentation Definitions~~

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

~~2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one~~

~~or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.~~

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

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

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

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

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

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

~~2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.~~

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

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

~~U. Deleted~~

~~1.0 DEFINITIONS~~

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

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

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

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

~~W. Deleted~~

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

~~Y. Surveillance Frequency – Relocated to Specifications 4.0.2 and 4.0.3.~~

~~Z. Surveillance Interval - Relocated to Specification 4.0.1.~~

~~AA. Deleted~~

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

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

~~DD. Deleted~~

~~EE. Deleted~~

~~FF. Deleted~~

~~GG. Deleted~~

~~HH. Deleted~~

~~II. Deleted~~

~~JJ. Deleted~~

~~KK. Deleted~~

~~LL. Deleted~~

~~MM. Deleted~~

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

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

3.0 LIMITING CONDITIONS FOR OPERATION APPLICABILITY

~~3.0.1 RESERVED~~

~~3.0.2 RESERVED~~

~~3.0.3 RESERVED~~

~~3.0.4 RESERVED~~

~~3.0.5 RESERVED~~

~~3.0.6 RESERVED~~

~~3.0.7 RESERVED~~

~~3.0.8 Inoperability of Snubbers~~

~~When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:~~

~~a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours;or~~

~~b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.~~

~~At the end of the specified period the required snubbers must be able to perform their associated support function(s) or the affected supported system LCO(s) shall be declared not met.~~

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 4.0.1

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

SR 4.0.2

Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. ~~The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.~~

SR 4.0.3

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

3.0 LIMITING CONDITIONS FOR OPERATION APPLICABILITY

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 4.0.3 (Continued)

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

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

~~BASES:~~

~~TS 3.0 Limiting Conditions for Operation Applicability~~

~~LCO 3.0.8 Bases~~

~~LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not being capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10CFR50.36(c)(2)(ii), and as such, are appropriate for control by the licensee.~~

~~If the allowed time expires and the snubbers(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the conditions and required actions entered.~~

~~LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.~~

~~LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.~~

~~LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10CFR50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.~~

TS 4.0 Surveillance Requirement (SR) Applicability

SR 4.0.1 Bases

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

~~Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:~~

~~a. The systems or components are known to be inoperable, although still meeting the SRs or~~
~~b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.~~

~~Surveillances do not have to be performed when the unit is in a mode or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.~~

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. ~~This allowance includes those SRs whose performance is normally precluded in a given mode or other specified condition.~~

~~Surveillances do not have to be performed on inoperable equipment because the LCOs define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 4.0.2, prior to returning equipment to OPERABLE status.~~

~~Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 4.0.2. Post maintenance testing may not be possible in the current SR 4.0.1 mode or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a mode or other specified condition where other necessary post maintenance tests can be completed.~~

~~An example of this process is:~~
~~a. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.~~

SR 4.0.2 Bases

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

The 25% extension does not significantly degrade the reliability that results from performing the surveillance at its specified frequency. This is based on the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 4.0.2 are those Surveillances for which the 25% extension of the interval specified in the frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. ~~An example of where SR 4.0.2 does not apply is in the Primary Containment Leakage Rate Testing Program. This program establishes testing requirements and frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.~~

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

SR 4.0.3 Bases

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

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

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements. ~~When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering Run Mode after each fuel loading, or in accordance with 10CFR50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 4.0.3 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. SR 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of Mode changes imposed by Action Statements.~~

SR 4.0.3 Bases (Continued)

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

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

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

3.1 3.1 RADIOACTIVE

~~3.8~~ LIMITING CONDITIONS FOR OPERATION

~~3.8~~ ~~RADIOATIVE~~ EFFLUENTS

Applicability:

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

Objective:

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

Specification:

~~A.~~ ~~Deleted~~

~~B.~~ ~~Deleted~~

~~C.~~ ~~Deleted~~

A ~~D~~. Liquid Holdup Tanks

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

2. With the quantity of radioactive material in any outside tank* exceeding the limit of Specification 3.8.D.1, immediately take action to suspend all additions of radioactive material to the tank. Within 48 hours, reduce the tank contents to within the limit.

4.1 4.1

~~4.8~~ SURVEILLANCE REQUIREMENTS

~~4.8~~ RADIOACTIVE EFFLUENTS

Applicability:

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

Objective:

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

Specification:

~~A.~~ ~~Deleted~~

~~B.~~ ~~Deleted~~

~~C.~~ ~~Deleted~~

A ~~D~~. Liquid Holdup Tanks

1. The quantity of radioactive material contained in each of the liquid holdup tanks* shall be determined to be within the limits of Specification 3.8.D.1 by analyzing a representative sample of the tank's contents within one week following the addition of radioactive materials to the tank. One sample may cover multiple additions.

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

~~3.8 LIMITING CONDITIONS FOR OPERATION~~

~~E. Deleted~~

~~F. Deleted~~

~~G. Deleted~~

~~H. Deleted~~

~~I. Deleted~~

~~J. Explosive Gas Mixture~~

~~1. If the hydrogen concentration in the off-gas downstream of the operating recombiner reaches four percent, take appropriate action that will restore the concentration to within the limit within 48 hours.~~

~~4.8 SURVEILLANCE REQUIREMENTS~~

~~E. Deleted~~

~~F. Deleted~~

~~G. Deleted~~

~~H. Deleted~~

~~I. Deleted~~

~~J. Explosive Gas Mixture~~

~~1. The concentration of hydrogen in the off-gas system downstream of the recombiners shall be continuously monitored by the hydrogen monitor required operable by the Offsite Dose Calculation Manual.~~

~~3.8 LIMITING CONDITIONS FOR OPERATION~~

~~K. Steam Jet Air Ejector (SJAE)~~

~~1. Gross radioactivity release rate from the SJAE shall be limited to less than or equal to 0.16 Ci/sec (after 30 minutes decay).~~

~~2. With the gross radioactivity release rate at the SJAE exceeding the above limit, restore the gross radioactivity release rate to within its limit within 72 hours or be in at least Hot Standby within the subsequent 12 hours.~~

~~3. With the gross radioactivity release rate at the SJAE greater than or equal to 1.5 Ci/sec (after 30-minute decay), restore the gross radioactivity release rate to less than 1.5 Ci/sec (after 30-minute decay), or be in Hot Standby within 12 hours.~~

~~L. Deleted~~

~~M. Deleted~~

~~N. Deleted~~

~~4.8 SURVEILLANCE REQUIREMENTS~~

~~K. Steam Jet Air Ejector (SJAE)~~

~~1. The gross radioactivity release rate shall be continuously monitored in accordance with the Offsite Dose Calculation Manual.~~

~~2. The gross radioactivity release rate of noble gases from the SJAE shall be determined to be within the limit of Specification 3.8.K.1 at the following frequencies by performing an isotopic analysis (for Xe-138, Xe-135, Xe-133, Kr-88, Kr-85m, Kr-87) on a representative sample of gases taken at the discharge.~~

~~a. Once per week.~~

~~b. Within 4 hours following an increase of 25% or 5000 microcuries/sec, whichever is greater, in steady-state activity levels during steady-state reactor operation, as indicated by the SJAE monitor.~~

~~L. Deleted~~

~~M. Deleted~~

~~N. Deleted~~

3.1

BASES:

~~3.8~~ ~~RADIOACTIVE~~ EFFLUENTS

~~A.~~ ~~Deleted~~

~~B.~~ ~~Deleted~~

~~C.~~ ~~Deleted~~

A ~~D~~. Liquid Holdup Tanks

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

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

~~E.~~ ~~Deleted~~

~~F.~~ ~~Deleted~~

~~G.~~ ~~Deleted~~

~~H.~~ ~~Deleted~~

~~I.~~ ~~Deleted~~

~~J.~~ ~~Explosive Gas Mixture~~

~~The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.~~

~~K.~~ ~~Steam Jet Air Ejector (SJAE)~~

~~Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total effective dose equivalent to an individual at the exclusion area boundary will not exceed the limits of 10CFR50.67 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.~~

~~BASES: 3.8 (Cont'd)~~

~~L. Deleted~~

~~M. Deleted~~

~~N. Deleted~~

~~Pages 177 through 189 have been intentionally deleted.~~

~~3.12~~ 3.2 LIMITING CONDITIONS FOR OPERATION

~~3.12~~ 3.2 ~~REFUELING AND~~ SPENT FUEL ~~HANDLING~~ STORAGE

Applicability:

Applies to ~~fuel handling, core reactivity limitations, and spent fuel handling.~~ storage of spent fuel.

Objective:

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

Specification:

~~A. Refueling Interlocks~~

~~The reactor mode switch shall be locked in the "Refuel" position during core alterations and;~~

~~1. The refueling interlocks shall be operable during in-vessel fuel movement for the equipment utilized in moving fuel.~~

~~If one or more of the required refueling interlocks are inoperable;~~

~~Immediately suspend fuel movement with equipment associated with the inoperable interlock(s),~~

~~-or-~~

~~Immediately insert a control rod withdrawal block and verify all control rods are fully inserted.~~

~~2. The refueling interlocks shall be operable except as specified in Specification 3.12.D and 3.12.E.~~

~~4.12~~ 4.2 SURVEILLANCE REQUIREMENTS

~~4.12~~ 4.2 ~~REFUELING AND~~ SPENT FUEL ~~HANDLING~~ STORAGE

Applicability:

Applies to the ~~periodic testing of those interlocks and instruments used during refueling.~~ parameters which monitor the storage of spent fuel.

Objective:

To verify ~~the operability of instrumentation and interlocks used in refueling~~ that spent fuel is being stored safely.

Specification:

~~A. Refueling Interlocks~~

~~Prior to any fuel handling, with the Head off the reactor vessel, the following required refueling interlock inputs shall be functionally tested once every 7 days:~~

~~a. All-rods-in;~~
~~b. Refuel platform position;~~
~~c. Refuel platform fuel grapple, fuel loaded;~~
~~d. Refuel platform frame mounted hoist, fuel loaded;~~
~~e. Refuel platform monorail mounted hoist, fuel loaded.~~

~~3.12 LIMITING CONDITIONS FOR OPERATION~~

~~B. Core Monitoring~~

~~During core alterations two SRMs shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable the following conditions shall be satisfied:~~

1. ~~The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detectors are connected into the proper circuitry which contain the required rod blocks).~~
2. ~~The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.~~

~~4.12 SURVEILLANCE REQUIREMENTS~~

~~B. Core Monitoring~~

~~Prior to making any alterations to the core the SRMs shall be functionally tested and checked for neutron response. Thereafter, the SRMs shall be checked daily for response.~~

3.12 LIMITING CONDITIONS FOR OPERATION

~~3.~~ ~~Prior to spiral unloading, the SRMs shall be proven operable as stated in Sections 3.12.B.1 and 3.12.B.2 above, however, during spiral unloading the count rate may drop below 3 cps.~~

~~4.~~ ~~Prior to spiral reloading, two diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, shall be loaded into core positions next to each of the 4 SRMs to obtain the required 3 cps. Until these eight bundles have been loaded, the 3 cps requirement is not necessary.~~

A ~~C~~. Fuel Storage Pool Water Level

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

4.12 SURVEILLANCE REQUIREMENTS

~~Prior to spiral unloading or reloading, the SRMs shall be functionally tested. Prior to spiral reloading, the SRMs shall be checked for neutron response.~~

A ~~C~~. Fuel Storage Pool Water Level

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

~~3.12 LIMITING CONDITIONS FOR OPERATION~~

~~D. Control Rod and Control Rod Drive Maintenance~~

~~One control rod may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance provided the following conditions are satisfied:~~

1. ~~The reactor mode switch shall be locked in the "Refuel" position and the required refueling interlocks shall be operable.~~

2. ~~Specification 3.3.A.1 shall be met, or the control rod directional control valves for a minimum of eight control rods surrounding the drive out of service for maintenance shall be disarmed and sufficient margin to criticality demonstrated.~~

3. ~~SRMs shall be operable in the core quadrant containing the control rod on which maintenance is being performed and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.12.B.~~

~~4.12 SURVEILLANCE REQUIREMENTS~~

~~D. Control Rod and Control Rod Drive Maintenance~~

1. ~~Prior to performing this maintenance, core shutdown margin shall be determined in accordance with Specification 3.3.A.1 to ensure that the core can be made subcritical at any time during the maintenance with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.~~

2. ~~Alternately, if a minimum of eight control rods surrounding the control rod out of service for maintenance are to be fully inserted and have their directional control valves electrically disarmed, the required shutdown margin shall be met with the strongest control rod remaining in service during the maintenance period fully withdrawn.~~

~~3.12 LIMITING CONDITIONS FOR OPERATION~~

~~E. Extended Core Maintenance~~

~~One or more control rods may be withdrawn or removed from the reactor core provided the following conditions are satisfied:~~

~~1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. The required refueling interlocks shall be operable.~~

~~2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.12.B.~~

~~3. If the spiral unload/reload method of core alteration is to be used, the following conditions shall be met:~~

~~a. Prior to spiral unload and reload, the SRMs shall be proven operable as stated in Specification 3.12.B1 and 3.12.B2. However, during spiral unloading, the count rate may drop below 3 cps.~~

~~4.12 SURVEILLANCE REQUIREMENTS~~

~~E. Extended Core Maintenance~~

~~Prior to control rod withdrawal for extended core maintenance, that control rod's control cell shall be verified to contain no fuel assemblies.~~

~~1. This surveillance requirement is the same as that given in Specification 4.12.A.~~

~~2. This surveillance requirement is the same as that given in Specification 4.12.B.~~

~~3.12 LIMITING CONDITIONS FOR OPERATION~~

~~4.12 SURVEILLANCE REQUIREMENTS~~

~~b. The core may be spirally reloaded to either the original configuration or a different configuration in the reverse sequence of that used to unload, with the exception that two (2) diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, shall be loaded into core positions next to each of the four (4) SRMs to obtain the required 3 cps. Until these eight (8) bundles have been loaded, the 3 cps requirement is not necessary. Following insertion of the initial eight (8) bundles, the reactor will be spirally reloaded around an SRM until the core is fully loaded.~~

~~c. At least 50% of the fuel assemblies to be reloaded into the core shall have previously accumulated a minimum exposure of 1000 Mwd/T.~~

~~3.12 LIMITING CONDITIONS FOR OPERATION~~

~~F.~~ ~~Fuel Movement~~

~~The reactor shall be shut down for a minimum of 24 hours prior to fuel movement within the reactor core.~~

~~G.~~ ~~Deleted~~

~~4.12 SURVEILLANCE REQUIREMENTS~~

~~F.~~ ~~Fuel Movement~~

~~Prior to any fuel handling or movement in the reactor core, the licensed operator shall verify that the reactor has been shut down for a minimum of 24 hours.~~

~~G.~~ ~~Deleted~~

~~3.12 LIMITING CONDITIONS FOR OPERATION~~

~~4.12 SURVEILLANCE REQUIREMENTS~~

B ~~H~~. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is stored in the spent fuel pool, the pool water temperature shall be maintained below 150°F.

B ~~H~~. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be recorded daily. If the pool water temperature reaches 150°F, all ~~refueling~~ operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.

3.2 & 4.2

SPENT FUEL STORAGE

BASES:

~~3.12 & 4.12~~ ~~REFUELING~~

~~A. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.~~

~~To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. Should the interlocks be made or found to be inoperable, the specifications offer an alternative to the cessation of fuel movement, not withstanding the completion of movement of a component to a safe position. The alternative is to immediately block control rod withdrawal and then perform a verification that all control rods are fully inserted. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the Reactivity Control System and interlocks at any time during shutdown or the following operating cycle.~~

~~The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist.~~

~~Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position, only one control rod can be withdrawn.~~

~~As discussed above, the purpose of the refueling interlocks is to prevent inadvertent criticality by ensuring that fuel is not loaded into a cell with a withdrawn control rod. The alternative identified within the specifications to continue fuel movement with inoperable interlocks satisfies this goal. The first refueling interlock safety function is to block control rod withdrawal whenever fuel is being moved in the reactor vessel. The alternative performs this function by requiring that a control rod block be placed in effect. The second refueling interlock safety function is to prevent fuel from being loaded into the vessel when a control rod is withdrawn. This function will continue to be performed by the second step of the alternative which is to verify that all control rods are fully inserted. Therefore, the alternative provides equal assurance against inadvertent criticality during fuel handling within the reactor vessel with inoperable interlocks.~~

~~The Surveillance Requirements for the refueling interlocks identify that the "required interlock inputs" shall be functionally tested. The intent of this statement is that only the interlock inputs associated with the equipment actually used to facilitate the core alteration is required to be functionally tested. For example, if the main mast is to be used for fuel movement, then the interlock inputs associated with the main mast need to be functionally tested. Conversely, if the frame mounted hoist and monorail mounted hoist, will not be utilized, then the interlock inputs associated with the frame mounted hoist and monorail mounted hoist need not be functionally tested.~~

BASES: ~~3.12 & 4.12 (Cont'd)~~

~~B.~~ ~~The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRMs will drop below 3 cps before all the fuel is unloaded. Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRMs will no longer be required. Requiring the SRMs to be operational prior to fuel removal assures that the SRMs are operable and can be relied on even when the count rate may go below 3 cps.~~

~~Prior to spiral reload, two diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, will be loaded into core positions next to each of the 4 SRMs to obtain the required 3 cps. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photo fission, and photo disintegration of deuterium in the moderator. This neutron production is normally great enough to meet the 3 cps minimum SRM requirement, thereby providing a means by which SRM response may be demonstrated before the spiral reload begins. During the spiral reload, the fuel will be loaded in the reverse sequence that it was unloaded with the exception of the initial eight (8) fuel assemblies which are loaded next to the SRMs to provide a means of SRM response.~~

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

~~D.~~ ~~During certain periods, it is desirable to perform maintenance on a single control rod and/or control rod drive. This specification provides assurance that inadvertent criticality does not occur during such maintenance.~~

~~The maintenance is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling operations as explained in Part A of these Bases. Refueling interlocks restrict the movement of control rods and the operation of the refueling equipment to reinforce operational procedures that prevent the reactor from becoming critical during refueling operations. During refueling operations, no more than one control rod is permitted to be withdrawn from a core cell containing one or more fuel assemblies. The refueling interlocks use the "full-in" position indicators to determine the position of all control rods. If the "full-in" position signal is not present for every control rod, then the "all-rods-in" permissive for the refueling equipment interlocks is not present and fuel loading and control rod withdrawal is prevented. The refuel position one-rod-out interlock will not allow the withdrawal of a second control rod. The requirement that an adequate shutdown margin be determined with the control rods remaining in service ensures that inadvertent criticality cannot occur during this maintenance. Disarming the directional control valves does not inhibit control rod scram capability.~~

BASES: ~~3.12 & 4.12 (Cont'd)~~

~~E.~~ ~~The intent of this specification is to permit the unloading of a portion of the reactor core for such purposes as inservice inspection requirements, examination of the core support plate, control rod, control rod drive maintenance, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.~~

~~This operation is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in the Bases for Specification 3.12.A. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.~~

~~One method available for unloading or reloading the core is the spiral unload/reload. Spiral reloading and unloading encompass reloading or unloading a cell on the edge of a continuous fueled region (the cell can be reloaded or unloaded in any sequence.) The pattern begins (for reloading) and ends (for unloading) around a single SRM. The spiral reloading pattern is the reverse of the unloading pattern, with the exception that two diagonally adjacent bundles, which have previously accumulated exposure in-core, and placed next to each of the four SRMs before the actual spiral reloading begins. The spiral reload can be to either the original configuration or a different configuration.~~

~~Additionally, at least 50% of the fuel assemblies to be reloaded into the core shall have previously accumulated a minimum exposure of 1000 Mwd/T to ensure the presence of a minimum neutron flux as described in Bases Section 3.12.B.~~

~~F.~~ ~~The intent of this specification is to assure that the reactor core has been shut down for at least 24 hours following power operation and prior to fuel handling or movement. The safety analysis for the postulated refueling accident assumed that the reactor had been shut down for 24 hours for fission product decay prior to any fuel handling which could result in dropping of a fuel assembly.~~

~~G.~~ ~~Deleted~~

B ~~H~~. The Spent Fuel Pool Cooling System is designed to maintain the pool water temperature below 125°F during normal ~~refueling~~ operations. If the reactor core is completely discharged, the temperature of the pool water may increase to greater than 125°F. ~~The RHR System supplemental fuel pool cooling may be used under these conditions to maintain the pool water temperature less than 150°F.~~

5.0 DESIGN FEATURES

5.1 Site

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

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

~~5.2 Reactor~~

~~A. The core shall consist of not more than 368 fuel assemblies.~~

~~B. The reactor core shall contain 89 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) or hafnium, or a combination of the two.~~

~~5.3 Reactor Vessel~~

~~The reactor vessel and applicable design codes shall be as described in Section 4 of the FSAR.~~

~~5.4 Containment~~

~~A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the FSAR.~~

~~B. The secondary containment shall be as described in subsection 5.3 of the FSAR and the applicable codes shall be as described in Section 12.0 of the FSAR.~~

~~C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in subsection 5.2 of the FSAR.~~

2

5.~~5~~ Spent ~~and New~~ Fuel Storage

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

A

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

B

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

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

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

BASES:

5.1 SITE

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

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

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

Note: Proposed changes to TS Section 6.0, with the exception of page renumbering, are under NRC review. Reference: Letter, Entergy Nuclear Operations, Inc., to NRC, "Technical Specifications Proposed Change No. 307 Revision to Mitigation Strategy License Condition and Technical Specification Administrative Controls for Permanently Defueled Condition," BVY 13-096, dated October 31, 2013 (ML13316A004)

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The plant manager ... ion and shall delegate ... ility during absences.

B. The plant manager ... implementation, ea ... n to systems or equipme ... hift

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~~ storage and maintenance of the ~~plant~~ nuclear fuel.

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

6. Deleted ~~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~~.

8. Deleted ~~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)

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

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. ~~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:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

d. In those cases where option (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

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

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

6.6 REPORTING REQUIREMENTS

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

A. Deleted

B. Deleted

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).~~

~~Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Responses to Request for Additional Information on FROSSTEY-2 Fuel Performance Code," BVY 91-024, dated March 6, 1991 (Approved by NRC SER, dated September 24, 1992).~~

~~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. <u>OFF-SITE DOSE CALCULATION MANUAL (ODCM)</u>

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.

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

Deleted

~~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;

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

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

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

E. TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

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

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

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

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

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

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

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

<u>Attachment 3</u>

Vermont Yankee Nuclear Power Station

Proposed Change 309

Retyped Operating License and Technical Specification Pages

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

(Vermont Yankee Nuclear Power Station)

Docket No. 50-271

Renewed Facility Operating License

Renewed Operating License No. DPR-28

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

a. This paragraph deleted by Amendment No. xxx.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

E. Entergy Nuclear Operations, Inc., pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. This paragraph deleted by Amendment No. xxx.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

C. Reports

Entergy Nuclear Operations, Inc. shall make reports in accordance with the requirements of the Technical Specifications.

D. This paragraph deleted by Amendment No. 226.

E. Environmental Conditions

Pursuant to the Initial Decision of the presiding Atomic Safety and Licensing Board issued February 27, 1973, the following conditions for the protection of the environment are incorporated herein:

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

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

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

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

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

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

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

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

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

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

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

    (b) Total curie alpha activity discharged.

    (c) Total curies of tritium discharged.

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

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

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

(g) Average concentration at discharge outfall.

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

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

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

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

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

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

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

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

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

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

F. This paragraph deleted by Amendment No. xxx.

G. Security Plan

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

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

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

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

J. License Transfer Conditions

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

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

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

a. Decommissioning Trust

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

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

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

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

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

b. Surety

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

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

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

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

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

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

K. This paragraph deleted by Amendment No. xxx.

L. This paragraph deleted by Amendment No. xxx.

M. This paragraph deleted by Amendment No. xxx.

N. This paragraph deleted by Amendment No. xxx.

O. This paragraph deleted by Amendment No. xxx.

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. This paragraph deleted by Amendment No. xxx.

S. This paragraph deleted by Amendment No. xxx.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
Eric J. Leeds

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosures:
Appendix A - Technical Specifications

Date of Issuance: March 21, 2011

APPENDIX A

TO

OPERATING LICENSE DPR-28

TECHNICAL SPECIFICATIONS

AND BASES

FOR

VERMONT YANKEE NUCLEAR POWER STATION

VERNON, VERMONT

ENTERGY NUCLEAR OPERATIONS, INC.

AND

ENTERGY NUCLEAR VERMONT YANKEE, LLC

DOCKET NO. 50-271

TABLE OF CONTENTS

1.0 DEFINITIONS

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

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

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

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

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

3.0 LIMITING CONDITIONS FOR OPERATION APPLICABILITY

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 4.0.1

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

SR 4.0.2

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

SR 4.0.3

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

3.0 LIMITING CONDITIONS FOR OPERATION APPLICABILITY

4.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 4.0.3 (Continued)

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

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

TS 4.0 Surveillance Requirement (SR) Applicability

SR 4.0.1 Bases

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

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

SR 4.0.2 Bases

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

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

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

SR 4.0.3 Bases

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

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

SR 4.0.3 Bases (Continued)

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

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

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

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

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 RADIOACTIVE EFFLUENTS

Applicability:

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

Objective:

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

Specification:

A. Liquid Holdup Tanks

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

2. With the quantity of radioactive material in any outside tank* exceeding the limit of Specification 3.8.D.1, immediately take action to suspend all additions of radioactive material to the tank. Within 48 hours, reduce the tank contents to within the limit.

4.1 SURVEILLANCE REQUIREMENTS

4.1 RADIOACTIVE EFFLUENTS

Applicability:

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

Objective:

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

Specification:

A. Liquid Holdup Tanks

1. The quantity of radioactive material contained in each of the liquid holdup tanks* shall be determined to be within the limits of Specification 3.8.D.1 by analyzing a representative sample of the tank's contents within one week following the addition of radioactive materials to the tank. One sample may cover multiple additions.

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

BASES:

3.1 RADIOACTIVE EFFLUENTS

A. Liquid Holdup Tanks

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

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

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 SPENT FUEL STORAGE

Applicability:

Applies to storage of spent fuel.

Objective:

To assure safe storage of spent fuel.

Specification:

A. Fuel Storage Pool Water Level

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

B. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is stored in the spent fuel pool, the pool water temperature shall be maintained below 150°F.

4.2 SURVEILLANCE REQUIREMENTS

4.2 SPENT FUEL STORAGE

Applicability:

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

Objective:

To verify that spent fuel is being stored safely.

Specification:

A. Fuel Storage Pool Water Level

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

B. Spent Fuel Pool Water Temperature

Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be recorded daily. If the pool water temperature reaches 150°F, all operations tending to raise the pool water temperature shall cease and measures taken immediately to reduce the pool water temperature below 150°F.

BASES:

3.2 & 4.2 SPENT FUEL STORAGE

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

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

5.0 DESIGN FEATURES

5.1 Site

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

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

5.2 Spent Fuel Storage

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

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

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

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

BASES:

5.1 SITE

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

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

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

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

A. The plant manager shall be responsible for overall 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.

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

6.2 ORGANIZATION (Cont'd)

B. 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

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.

6.4 PROCEDURES (Cont'd)

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:

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

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

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

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

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

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

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

6.5 HIGH RADIATION AREA (Cont'd)

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

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

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

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

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

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

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

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

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

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

6.5 HIGH RADIATION AREA (Cont'd)

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

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

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

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

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

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

   d. In those cases where option (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

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

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

6.6 REPORTING REQUIREMENTS

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

A. Deleted

B. Deleted

C. Deleted

D. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the 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.

6.7 PROGRAMS AND MANUALS (Cont'd)

1. Licensee initiated changes to the ODCM:

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

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

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

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

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

C. Deleted

D. Radioactive Effluent Controls Program

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

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

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

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

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

6.7 PROGRAMS AND MANUALS (Cont'd)

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

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

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

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

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

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

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

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

E. TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

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

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

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

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

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

6.7 PROGRAMS AND MANUALS (Cont'd)

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

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