Entergy

Entergy Nuclear Operations, Inc.
Vermont Yankee
320 Governor Hunt Rd
Vernon, VT 05354
Tel 802 257 7711

Christopher J. Wamser
Site Vice President

BVY 14-036

June 9, 2014

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

SUBJECT: Technical Specifications Proposed Change No. 306 Eliminate Certain ESF Requirements during Movement of Irradiated Fuel - Supplement 1 (TAC No. MF3068)
Vermont Yankee Nuclear Power Station
Docket No. 50-271
License No. DPR-28

REFERENCES:

1. 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 (TAC No. MF3068) (ADAMS Accession No. ML13323A516)

2. Email, USNRC to Entergy Nuclear Operations, Inc. "Vermont Yankee RAI for LAR on Eliminate Certain ESF Requirements During Movement of Irradiated Fuel (TAC No. MF3068)," dated May 19, 2014

Dear Sir or Madam:

By letter dated November 14, 2013 (Reference 1), Entergy Nuclear Operations, Inc. (ENO) proposed an amendment to Renewed Facility Operating License (OL) DPR-28 for Vermont Yankee Nuclear Power Station (VY). The proposed amendment would change the Technical Specification (TS) requirements associated with handling irradiated fuel and performing core alterations. 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.

In Reference 2, the NRC provided VY with a Request for Additional Information (RAI) regarding the proposed changes. Attachment 1 of this letter provides the responses to the RAI. Attachment 2 of this letter provides a revised markup of the VY TS pages affected by the RAI response.

The conclusions of the no significant hazards consideration and the environmental considerations contained in Reference 1 are not affected by, and remain applicable to, this supplement.

This letter contains no new regulatory commitments.

A001
NRR

If you have any questions on this transmittal, please contact Mr. Philip Couture at 802-451-3193.

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

Executed on June 9, 2014.

Sincerely,

CJW/plc

Attachments: 1. Response to Request for Additional Information
2. Markup of Technical Specification Pages

cc: Mr. William M. Dean
Region 1 Administrator
U.S. Nuclear Regulatory Commission
2100 Renaissance Blvd, Suite 100
King of Prussia, PA 19406-2713

Mr. James S. Kim, Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O8D15
Washington, DC 20555

USNRC Resident Inspector
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
Vernon, VT 05354

Mr. Christopher Recchia, Commissioner
VT Department of Public Service
112 State Street, Drawer 20
Montpelier, VT 05620-2601

Attachment 1

Vermont Yankee Nuclear Power Station

Response to Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION TASK FORCE TRAVELER 51
VERMONT YANKEE NUCLEAR POWER STATION
DOCKET NO. 50-271

By application dated November 14, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13323A518), Entergy Nuclear Operations submitted a license amendment for Vermont Yankee (VY). The proposed license amendment request (LAR) would eliminate operability requirements for secondary containment when handling sufficiently decayed irradiated fuel and while performing core alterations using Technical Specification Task Force (TSTF) – 51, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations."

RAI 1

Attachment 4, Table 3-2, entitled "VYNPP [VY Nuclear Power Plant] – Re-analysis of AST/FHA [alternative source term/fuel handling accident] Radiological Consequences with Open Containment" (ADAMS Accession No. ML13323A519) of the November 14, 2013 application, provides a core inventory based upon a core average maximum burnup of 58 giga-watt-days per metric ton of uranium (GWD/MTU). Attachment 4, Table 3-1 states that the FHA uses Table 3 gap fractions from Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Adams Accession Number ML003716792). Footnote 11 for Table 3 of RG 1.183 states that Table 3 is acceptable for use with currently approved reactor light water fuel with a peak burnup of up to 62,000 mega-watt-days per metric ton of uranium (MWD/MTU) (equivalent to 62 GWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatts per foot (kW/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU (equivalent to 54 GWD/MTU). Since the assumed fuel burnup for the Attachment 4, Table 3-2 core inventories appear to exceed the RG 1.183, footnote 11 limits, please confirm that the VY fuel burnup and linear heat generation rates comply with footnote 11. If not, please justify the use of Table 3 from RG 1.183 with fuel outside the burnup and linear heat generation rates used to derive Table 3.

Response

Entergy Nuclear Operations, Inc. (ENO) confirms that the fuel burnup and linear heat generation rates comply with the RG 1.183 footnote 11 limits on burnup and maximum linear heat generation rate. The RG 1.183 footnote 11 limits were specifically evaluated for the current VY operating cycle during reload licensing by Global Nuclear Fuel (GNF). Only significantly different control rod patterns and operation could result in exceeding the 6.3 kW/ft linear heat generation rate for those fuel rods with burnup greater than 54 GWD/MTU. Such changes would be evaluated by ENO and GNF before implementation.

The RG 1.183 footnote 11 limits were again evaluated by GNF for the planned extended operating cycle prior to the final shutdown of VY and were likewise found to be met. ENO reviewed the verified summary edits from GNF and found that the limits were met through the end of the planned cycle with significant margin.

RAI 2

Page 9 of 17 of the application, entitled "Technical Specifications Proposed Change No. 306, Eliminate Certain ESF [Engineered Safety Feature] Requirements during Movement of Irradiated Fuel," dated November 14, 2013 (ADAMS Accession Number ML13323A518) states:

> ***The accidents postulated to occur during core alterations, in addition to the fuel handling accident, are* [the] *inadvertent criticality due to control rod removal error and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Therefore, the only accident postulated to occur during core alterations that result in significant radioactive release is the FHA* [fuel handling accident]. *Thus, the consequence of a FHA envelops the consequences of potential accidents postulated to occur during core alterations.***

Page 14 of 17 of the application also states that the proposed changes follow Technical Specification Task Force traveler 51 (TSTF-51), Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations" (ADAMS Accession Number ML040400343).

TSTF-51 states:

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

Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (ADAMS Accession Number ML003734190) states:

> ***The models, assumptions, and parameter inputs used by the licensee should be reviewed to ensure that the conservative design basis assumptions outlined in RG-1.183 have been incorporated.***

Appendix B of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession Number ML003716792), Regulatory Position 1.1 states:

> ***The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case.***

After reviewing the information submitted by the VY submittal to adopt TSTF-51, the NRC staff needs additional information to verify that the limiting cases have been considered.

a. **Please provide a FHA analysis that evaluates the dropping of loads allowed over irradiated fuel assemblies (i.e. sources or reactivity control components) onto irradiated fuel assemblies with 24-hours of decay time. The analysis should only credit those safety systems required to be operable as required by technical**

specification [TS]. This will provide the staff with reasonable assurance that the FHA doses remain within regulatory limits when references to Core Alterations are removed from TSs and ESFs are no longer required during movement of loads such as sources or reactivity control components.

b. **Page 7 of 17 of the application dated November 14, 2013 states that two main configurations of the Reactor Building during fuel movement were considered. The second configuration discusses "various [emphasis added] pre- and post-FHA Main Control Room (CR) ventilation configurations that would support refueling with open containment," but does not define which configurations are credited in the proposed TS changes. These ventilation configurations are discussed in the submittal, but the NRC staff needs some clarification regarding these configurations. Please state the proposed new design basis configuration credited to support the TS changes.**

Response

a. To address the NRC staff concerns with removal of references to Core Alterations from the TS, ENO is retracting from the license amendment request (Reference 1) the proposed removal of any references to Core Alterations from the VY TS. This retraction is based on the planned permanent cessation of power operations of VY at the end of the current operating cycle, which is expected to occur in the fourth quarter of 2014 (Reference 2). Once the VY reactor is permanently defueled and the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are docketed 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 and the term Core Alteration will have no meaning as there will no longer be a reactivity concern with the reactor core. Revised markups to the affected TS pages are provided in Attachment 2 of this letter. As in the LAR, proposed changes to the TS Bases are provided for information 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.

   To provide additional assurance that the changes proposed in Reference 1 will not be implemented while there is still fuel in the VY reactor vessel or before 13 days have passed following permanent cessation of operations, ENO is revising the requested approval date of the proposed changes to be contingent upon the docketing of the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(i) and (ii) and following a minimum of 13 days after the permanent cessation of operations. ENO no longer requests that the proposed changes be approved by December 1, 2014.

b. No new design basis configuration is proposed as part of TS changes. The CR ventilation configuration credited to support the proposed TS changes does not differ from the current VY design associated with the analysis of record. The analysis supporting the proposed TS change utilizes the following inputs:

   - No credit for station containment systems (i.e. "open" containment)
   - Release point to atmosphere is via a Reactor Building (RB) blowout panel (ground level release)
   - Release duration to atmosphere is 2 hour release duration per RG 1.183, Appendix B
   - 3,700 cfm unfiltered Control Room intake
   - 30 day exposure interval for the Control Room per conservative assumption

- 2 hour exposure interval for the Exclusion Area Boundary (EAB) and Low Populations Zone (LPZ) per RG 1.183, Section 4.1.5

Alternate configurations discussed in the FHA analysis are not credited to support the proposed TS changes. These cases were run for sensitivity studies not associated with the proposed changes.

Also note that, as discussed in Section 2.2.4 of the VY Updated Final Safety Analysis Report (UFSAR), the EAB is 910 feet from the reactor at the closest point and the LPZ is a 5 mile radius and accordingly the dose at the LPZ would be lower. As stated in the current analysis of record, calculation of the LPZ dose was considered not to be necessary because the EAB dose is more limiting.

RAI 3

Page 10 of 17 of the application, dated November 14, 2013, states:

The operability requirements during movement of a fuel cask for ESF mitigation are deleted as part of this proposed license amendment.

and,

Since the FHA resulting from a dropped fuel cask is shown to not be credible, the proposed TS changes omitting operability requirements during movement of a fuel cask ESF mitigation is justified.

SRP 15.7.5, "Spent Fuel Cask Drop Accidents," (ADAM Accession No. ML052350315) states:

A design basis radiological analysis is performed if a cask drop exceeding 30 feet can be postulated or if limiting devices are removed during cask handling within the plant so the 30-foot drop height is exceeded. If the radiological consequences of a cask drop accident are to be computed, then information on whether building leaktightness can be expected after a cask drop is obtained from ASB **[Auxiliary Systems Branch]** ***(e.g., whether the technical specifications require large doors to be closed during fuel handling or whether ventilation systems should be operating and whether the building leaktightness would be violated by the cask drop).***

At VY can a spent fuel cask drop exceed 30 feet or can the limiting devices be removed during cask handling? If so, please provide the radiological consequences of a cask drop accident. Please justify all answers.

Response

While a spent fuel storage cask can be raised to a height exceeding 30 feet, a cask drop is not a postulated event at VY. The basis for this determination is documented in Section 3.6 of the LAR (Reference 1).

RAI 4

Page 9 of Attachment 4 of the application, dated November 14, 2013, states that the activity releases from the containment atmosphere over two hours is 98.2%. Appendix B of RG 1.183, Regulatory Position 5.3 states if the containment is open during fuel handling, the radioactivity that escapes from the reactor cavity pool to the containment is released to the environment over a two-hour time period. RG 1.83, Regulatory Position 5.1.2, "Assignment

of Numeric Input Values," states that the numeric values that are chosen as inputs to the required analysis should be selected with the objective of determining a conservative dose. Please justify why a conservative value of 100% of the activity in containment was not assumed to be released from the containment over the two hour time period.

Response

As described in Section 2.2 of Attachment 4 of the LAR (Reference 1), an exponential release to the environment was assumed for the post-FHA radioactive material that escapes the water pool, based on a building air exchange rate of 2.0 air changes per hour. This air exchange rate leads to [1.0 - exp{-2.0 (hr^{-1}) * 2 (hr)}] = 98.17% of the airborne activity within the reactor building getting released within 2 hrs. This analytical approach using the air exchange rate was selected as a reasonable conservative assumption employed in the calculation to support the proposed changes.

The table below provides updated EAB dose rates for 100% release of the source within a 2 hour interval. This is based on simply increasing the doses documented in Table 5-3 of Attachment 4 of the LAR by (100 – 98.17) = 1.83%. It is seen that the regulatory limit of 6.3 rem is met in all cases.

VY FHA WITH OPEN CONTAINMENT - EAB DOSE

Decay Time (days)	EAB TEDE Dose (rem)	
	Original (98.17% Release)	Adjusted (100% Release)
1	5.895	6.003[(a)]
3	3.643	3.710
5	2.953	3.007
7	2.451	2.496
9	2.042	2.079
11	1.705	1.736
13	1.424	1.450
15	1.190	1.212
17	0.9957	1.014
19	0.8333	0.8485

(a) 5.895 x 1.0183 = 6.003 rem

The CR doses reported in Table 5-3 of Attachment 4 of the LAR do not require a similar adjustment since the dose analysis is for a period of 30 days. This is clarified in Section 2.2 of Attachment 4 of the LAR, which states:

> *It is noted that, for the MCR 30-day dose computations, the releases from the RB were assumed to continue for 30 days. Included in the releases beyond 2 hours are the (100 – 98.2) = 1.8% still airborne within the RB at 2 hrs, as well as the noble gases generated by the decay of iodines retained by the pool water.*

There is no change to the 13 day time period after shutdown required for the fuel to decay such that the dose limits are not exceeded, since the CR dose is not impacted by this added conservatism and this dose was the limiting factor in the determination of the time period.

RAI 5

Page 10 of Attachment 4 of the application, dated November 14, 2013, states that four sensitivity cases make use of several rates to assess the dose impact on the main CR purge initiation time. Please describe which case is to be reviewed for the design basis and clarify what is meant by the "purge initiation time."

Response

See response to RAI 2.b. The term "purge initiation time' is associated with the non-design basis sensitivity study described in Table 5-6, Page 31 of Attachment 4 of the LAR that is not associated with the proposed TS changes.

RAI 6

Regulatory Position 5.1.2 of RG 1.183 states: "The single active component failure that results in the most limiting radiological consequences should be assumed." State the most limiting single active failure for FHA and justify the answer.

Response

There are no ESF components employed or credited in the FHA analysis, hence there was no requirement for single-failure assumption. No single failure is postulated. It is noted, in particular, that the CR ventilation system has no filtration capability, and that it was assumed to be in the normal operating mode.

RAI 7

Attachment 4, Table 3-1, dated November 14, 2013, states that VY assumes an overall pool decontamination factor (or DF) of 200 based upon Appendix B of RG 1.183. The DF of 200 is based upon reference B-1 ("Evaluation of Fission Product Release and Transport," (ADAMS Accession No. 8402080322)) of RG 1.183. The data upon which the pool DF of 200 is based was developed in 1971 and was based on the Westinghouse fuel marketed at the time (the assumed internal fuel pressure of 1200 pounds force per square inch gage (psig) was used). Since higher pressures correlate to lower DFs, the NRC staff would like VY to confirm that the fuel VY uses will have an internal fuel pressure of less than 1200 psig. If not, please provide the experimental data for current fuel types used at VY that justify a DF of 200 for fuel pressures greater than 1200 psig. Also, please provide a detailed justification for using a DF of 200 for pressures up to 1200 psig.

Response

ENO confirms that the fuel in use at VY has an internal pressure of less than 1200 psig.

RAI 8

Please provide a justification for all changes from the current licensing basis (See Issue 1 of NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," (ADAMS Accession No. ML053460347) for more detail). No justification is needed for changes that are consistent with Regulatory Guide 1.183 or are provided in the submittal dated August 13, 2013 (ADAMS Accession No. ML13247A076) unless requested by these RAIs.

Response

Justification for all changes to the current licensing basis was provided in the application for the license amendment, as supplemented by the responses to these RAIs.

VY has previously received NRC approval for full-scope implementation of AST by letter dated March 29, 2005 (Reference 3).

The analysis performed in support of the proposed changes and provided in the application was intended to answer the specific question of how long it would take for used reactor fuel to decay to the point that the radiological consequences of a FHA would not result in the offsite and control room accident dose criteria being exceeded. The analysis is not intended to supersede the AST-based FHA analysis that was previously reviewed by the NRC staff.

The analysis provided in the application addressed all characteristics of the AST related to the FHA and the TEDE criteria as described in the VY design basis. Therefore, the conclusions reached by the NRC staff in approving full implementation of AST at VY remain valid.

RAI 9

The changes to TS 3.9.4 allow an "open" containment when moving fuel that is not recently irradiated. Consistent with Regulatory Issue Summary 2006-04, please confirm that all pathways to the environment created by the proposed changes are considered and analyzed in the FHA analysis.

a. **Please confirm that the most limiting combination of release point and receptor for the control room were used to determine atmospheric dispersion factors for each accident.**

b. **State and justify the release points that correlate to the atmospheric dispersion factors used.**

Response

The release point and atmospheric dispersion factors employed in the FHA analysis with open containment are identified in Table 3-3 of Attachment 4 of the LAR. The release point is the RB blowout panels. This forms the most limiting combination when paired with the receptor for the main control room based on the proximity of the RB blowout panels to the main control room air intake. The atmospheric dispersion factors were based on ARCON-96, along with the following:

- 5 year's worth of hourly meteorological data collected on site (1995-1999).
- The building area used for the wake correction was the projected area of the reactor building wall facing the control room air intake.
- The distance from the source (mid-point of the RB blowout panel) to the control room air intake (32 feet), source/receptor elevations, and the wind direction were based on site drawings. The RB siding facing the Control Room intake was treated conservatively as a point source (USFAR, Table 14.6.10).

It is noted that the atmospheric dispersion factors used in Attachment 4 of the LAR are the same as those used in the applicable VY calculation of record for implementation of the AST methodology.

RAI 10

SRP 16.0, "Technical Specifications," (ADAMS Accession No. ML100351425) states: "In TS change requests for facilities with TS based on previous STS [Standard Technical Specifications], licensees should comply with comparable provisions in these STS NUREGs to the extent possible or justify deviations from the STS." Please provide a justification for deviations from the STS created by the proposed changes.

Response

The following provides a comparison of the proposed changes to the VY TS to the changes proposed to the STS for NUREG-1433 in TSTF-51A. NUREG-1433, Revision 4 is the STS for General Electric (GE) Boiling Water Reactor/4 (BWR/4) plants (Reference 4). VY is a GE BWR/4 plant. The following discussion also identifies any deviations from the proposed STS changes in TSTF-51A. It is noted here that VY has "custom" TS and has not performed a conversion to the NUREG-1433 STS. Therefore, there are inherent wording differences between the equivalent VY TS and STS.

TS Table 3.2.3, Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Instrumentation:

TS Table 3.2.3 contains the instrumentation equivalent to those listed in STS Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation. Footnotes (c) and (d), applicable to the High Reactor Building Ventilation Radiation (Trip Function 3) and High Refuel Floor Zone Radiation (Trip Function 4) trip functions are proposed for revision.

Footnote (c) is proposed to be revised to be consistent with footnote (b) of STS Table 3.3.6.2-1, with brackets removed from "recently" and "secondary". There is no deviation from the STS with the proposed change to footnote (c). The proposed removal of "or fuel cask" from footnote (c) is an additional change not covered by the scope of TSTF-51A.

Footnote (d) [During Alteration of the Reactor Core] was proposed for deletion to be consistent with the footnotes of STS Table 3.3.6.2-1 as there is no footnote related to Core Alterations in STS Table 3.3.6.2-1. Reference to Core Alterations was removed from footnote (c) of STS Table 3.3.6.2-1 by TSTF-51A. However, based on the response to RAI 2 of this letter, ENO is retracting the portions of the proposed changes that would eliminate references to Core Alterations, thereby creating a deviation from the STS.

TS 3.7.B.4, Standby Gas Treatment System:

Proposed TS 3.7.B.4.b was to be revised to be consistent with the APPLICABILITY section of STS 3.6.4.3, Standby Gas Treatment (SGT) System in that reference to Core Alterations is removed and "recently" is added in front of "irradiated fuel" with brackets removed from "recently" and "secondary." TSTF-51A removed the words "During CORE ALTERATIONS" and added "[recently]" in front of "irradiated fuel" in the APPLICABILITY section. The proposed removal of "or the fuel cask" from TS 3.7.B.4.b is an additional change not covered by the scope of TSTF-51A. Based on the response to RAI 2 of this letter, ENO is retracting the portions of the proposed changes that would eliminate references to Core Alterations, thereby creating a deviation from the STS. There is no deviation from the STS with the proposed changes to TS 3.7.B.4.b in terms of the addition of "recently" in front of "irradiated fuel."

Proposed TS 3.7.B.4.b.i will be revised to be consistent with STS 3.6.4.3 REQUIRED ACTION C.2.1 in that "recently" is added in front of "irradiated fuel" with brackets removed from "recently" and "secondary." The proposed removal of "or the fuel cask" from TS 3.7.B.4.b.i is an additional change not covered by the scope of TSTF-51A. There is no deviation from the STS with the proposed changes to TS 3.7.B.4.b.i.

Proposed TS 3.7.B.4.b.ii was proposed for deletion to be consistent with the required actions of STS 3.6.4.3 as reference to Core Alterations was removed from STS 3.6.4.3 CONDITION C by TSTF-51A. However, based on the response to RAI 2 of this letter, ENO is retracting the portions of the proposed changes that would eliminate references to Core Alterations, thereby creating a deviation from the STS.

TS 3.7.C, Secondary Containment System:

Proposed TS 3.7.C.1.b and TS 3.7.C.1.c were proposed to be revised to be consistent with the APPLICABILITY section of STS 3.6.4.1, Secondary Containment, in that reference to Core Alterations is removed from TS 3.7.C.1.c and, for TS 3.7.C.1.b, "recently" is added in front of "irradiated fuel" with brackets removed from "recently" and "secondary." TSTF-51A removed the words "During CORE ALTERATIONS" and added "[recently]" in front of "irradiated fuel" in the APPLICABILITY section. The proposed removal of "or the fuel cask" from TS 3.7.C.1.b is an additional change not covered by the scope of TSTF-51A. Based on the response to RAI 2 of this letter, ENO is retracting the portions of the proposed changes that would eliminate references to Core Alterations, thereby creating a deviation from the STS. There is no deviation from the STS with the proposed changes to TS 3.7.C.1.b.

Proposed TS 3.7.C.4 will be revised to be consistent with STS 3.6.4.1 CONDITION C in that reference to Core Alterations is removed from TS 3.7.C.4 and "recently" is added in front of "irradiated fuel" with brackets removed from "recently" and "secondary." The proposed removal of "or the fuel cask" from TS 3.7.C.4 is an additional change not covered by the scope of TSTF-51A. Based on the response to RAI 2 of this letter, ENO is retracting the portions of the proposed changes that would eliminate references to Core Alterations, thereby creating a deviation from the STS. There is no deviation from the STS with the proposed change to TS 3.7.C.4 in terms of the addition of "recently" in front of "irradiated fuel."

Proposed TS 3.7.C.4.a will be revised to be consistent with the STS 3.6.4.1 REQUIRED ACTION C.1, in that "recently" is added in front of "irradiated fuel" with brackets removed from "recently" and "secondary." TSTF-51A added "[recently]" in front of "irradiated fuel" in REQUIRED ACTION C.1. The proposed removal of "and the fuel cask" from TS 3.7.C.4.a is an additional change not covered by the scope of TSTF-51A. There is no deviation from the STS with the proposed changes to TS 3.7.C.4.a.

Proposed TS 3.7.C.4.b was proposed for deletion to be consistent with STS 3.6.4.1 REQUIRED ACTION C.2. TSTF-51A removed "Suspend CORE ALTERATIONS" from REQUIRED ACTION C.2. However, based on the response to RAI 2 of this letter, ENO is retracting the portions of the proposed changes that would eliminate references to Core Alterations, thereby creating a deviation from the STS.

TSTF-51A also included changes to the following STS sections applicable to NUREG-1433:

- AC Sources - Shutdown (STS 3.8.2)
- DC Sources - Shutdown (STS 3.8.5)
- Inverters - Shutdown (STS 3.8.8)
- Distribution Systems - Shutdown (STS 3.8.10)
- Secondary Containment Isolation Valves (STS 3.6.4.2)
- Main Control Room Environmental Control System Instrumentation (STS 3.3.7.1)
- Primary Containment Isolation Valves (STS 3.6.1.3)
- Main Control Room Environmental Control System (STS 3.7.4)
- Control Room Air Conditioning System (STS 3.7.5)

VY reviewed those STS sections for applicability to the VY TS and determined that no additional changes to the VY TS were required in order to implement the proposed changes. In particular TS 3.7.E (Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)) was not included in the scope of the proposed changes on the basis that this specification only applies when secondary containment integrity is required. TS 3.7.E is consistent with the applicable STS section, 3.6.4.2 (Secondary Containment Isolation Valves). Omission of TS 3.7.E from the proposed changes is acceptable because once VY dockets the certifications of permanent cessation of power operations and permanent defueling of the reactor required by 10 CFR 50.82(a)(1)(i) and (ii), respectively, pursuant to 10 CFR 50.82(a)(2), the Part 50 license will no longer authorize operation of the VY reactor or emplacement or retention of fuel within the VY reactor vessel. Since the VY reactor will not be allowed to be refueled and operated again, new "recently" irradiated fuel will not be able to be generated. Upon approval of the proposed changes to TS 3.7.C, secondary containment integrity will not be required once the required fuel decay time passes and, accordingly, it will no longer be possible for TS 3.7.E to be applicable.

The fact that TSTF-51A included changes to the STS that were determined to not be applicable to the VY TS does not by itself represent a deviation from the STS.

No justification for changes to the VY TS Bases based on corresponding changes made to the STS Bases by TSTF-51A is provided since the changes to the TS Bases were provided for information only and will be incorporated in accordance with TS 6.7.E, the VY TS Bases Control Program.

References:

1. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Technical Specifications Proposed Change No. 306 Eliminate Certain ESF Requirements during Movement of Irradiated Fuel (TAC No. MF3068)," BVY 13-097, dated November 14, 2013 (ML13323A518)
2. Letter, Entergy Nuclear Operations, Inc. to USNRC, "Notification of Permanent Cessation of Power Operations," BVY 13-079, dated September 23, 2013 (ML13273A204)
3. Letter, USNRC to Entergy Nuclear Operations, Inc., "Vermont Yankee Nuclear Power Station Issuance of Amendment RE: Alternative Source Term (TAC No. MC0253)," NVY 05-045, dated March 29, 2005
4. NUREG-1433, Standard Technical Specifications, General Electric BWR/4 Plants, Revision 4.0, published April 2012

Attachment 2

Vermont Yankee Nuclear Power Station

Markup of Technical Specification Pages

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

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

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

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

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

(d) During Alteration of the Reactor Core.

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

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

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

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

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

3.7 LIMITING CONDITIONS FOR OPERATION

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

ii. Suspend core alterations; and

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

C. Secondary Containment System

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

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

4.7 SURVEILLANCE REQUIREMENTS

C. Secondary Containment System

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

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

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

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

3.7 LIMITING CONDITIONS FOR OPERATION

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

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

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

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

   b. Suspend alteration of the Reactor Core; and

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

4.7 SURVEILLANCE REQUIREMENTS

2. Intentionally blank.

3. Intentionally blank.

4. Intentionally blank.

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

instrumentation are implicitly assumed in the safety analyses of References 2, 3, and 4, to initiate closure of the RBAVSIVs and start the SGT System to limit offsite doses.

Reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The operability of the reactor building ventilation isolation and Standby Gas Treatment System initiation instrumentation is dependent on the operability of the individual instrumentation channel Trip Functions specified in Table 3.2.3. Each Trip Function must have the required number of operable channels in each trip system, with their trip setpoints within the calculational as-found tolerances specified in plant procedures. Operation with actual trip setpoints within calculational as-found tolerances provides reasonable assurance that, under worst case design basis conditions, the associated trip will occur within the Trip Settings specified in Table 3.2.3. As a result, a channel is considered inoperable if its actual trip setpoint is not within the calculational as-found tolerances specified in plant procedures. The actual trip setpoint is calibrated consistent with applicable setpoint methodology assumptions.

In general, the individual Trip Functions are required to be OPERABLE in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature > 212°F), during operations with the potential for draining the reactor vessel (OPDRVs)~~,~~ and during movement of recently irradiated fuel assemblies ~~or fuel cask~~ in secondary containment, and during Alteration of the Reactor Core; consistent with the Applicability for the SGT System and secondary containment requirements in Specifications 3.7.B and 3.7.C. Trip Functions that have different Applicabilities are discussed below in the individual Trip Functions discussion.

Insert 1

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Trip Function by Trip Function basis.

1. Low Reactor Vessel Water Level

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite release. The Low Reactor Vessel Water Level Trip Function is one of the Trip Functions assumed to be operable and capable of providing isolation and initiation signals. The isolation and initiation of systems on Low Reactor Vessel Water Level support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Low Reactor Vessel Water Level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Low Reactor Vessel Water Level Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation and initiation function.

BASES: 3.2.C/4.2.C REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When High Reactor Building Ventilation Radiation or High Refueling Floor Zone Radiation is detected,

secondary containment isolation and actuation of the SGT System are initiated to support actions to limit the release of fission products as assumed in the UFSAR safety analyses (Ref. 4).

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation signals are initiated from radiation detectors that are located on the ventilation exhaust duct coming from the reactor building and the refueling floor zones, respectively. Two channels of High Reactor Building Ventilation Radiation Trip Function and two channels of High Refueling Floor Radiation Trip Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation and initiation function.

The Trip Settings are chosen to promptly detect gross failure of the fuel cladding.

The High Reactor Building Ventilation Radiation and High Refueling Floor Zone Radiation Trip Functions are required to be operable in RUN, STARTUP/HOT STANDBY, HOT SHUTDOWN, Refuel (with reactor coolant temperature > 212°F) where considerable energy exists in the RCS; thus, there is a possibility of pipe breaks resulting in significant releases of radioactive steam and gas. In COLD SHUTDOWN and Refuel (with reactor coolant temperature ≤ 212°F), the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these Modes; thus, these Trip Functions are not required. In addition, the Trip Functions are also required to be operable during OPDRVs~~,~~ and during movement of recently irradiated fuel assemblies ~~or fuel cask~~ in the secondary containment, and during Alteration of the Reactor Core, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

Insert 1

ACTIONS

Table 3.2.3 ACTION Note 1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours depending on the Trip Function (12 hours for those Trip Functions that have channel components common to RPS instrumentation, i.e., Trip Functions 1 and 2, and 24 hours for those Trip Functions that do not have channel components common to RPS instrumentation, i.e., all other Trip Functions), has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to operable status. This out of service time is only acceptable provided the associated Trip Function is still maintaining isolation capability (refer to next paragraph). If the inoperable channel cannot be restored to operable status within the allowable out of service time, the channel must be placed in the tripped condition per Table 3.2.3 Note 1.a.1) or 1.a.2), as applicable. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately,

BASES: 3.7 (Cont'd)

surveillances such as monthly torus to drywell vacuum breaker tests. Procedurally, when AC-6A is open, AC-6 and AC-7 are closed to prevent overpressurization of the SBGT system or the reactor building ductwork, should a LOCA occur. For this and similar analyses performed, a spurious opening of AC-6 or AC-7 (one of the closed containment isolation valves) is not assumed as a failure simultaneous with a postulated LOCA. Analyses demonstrate that for normal plant operation system alignments, including surveillances such as those described above, that SBGT integrity would be maintained if a LOCA was postulated. Therefore, during normal plant operations, the 90 hour clock does not apply. Accordingly, opening of the 18 inch atmospheric control isolation valves AC-7A, AC-7B, AC-8 and AC-10 will be limited to 90 hours per calendar year (except for performance of the subject valve stroke time surveillances - in which case the appropriate corresponding valves are closed to protect equipment should a LOCA occur). This restriction will apply whenever primary containment integrity is required. The 90 hour clock will apply anytime purge and vent evolutions can not assure the integrity of the SBGT trains or related equipment.

B. and C. Standby Gas Treatment System and Secondary Containment System

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. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading and low power physics testing.

In the Cold Shutdown condition or the Refuel Mode,the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these conditions. Therefore, maintaining Secondary Containment Integrity is not required in the Cold Shutdown condition or the Refuel Mode, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel, during alteration of the Reactor Core, or during movement of recently irradiated fuel assemblies ~~or the fuel cask~~ in the secondary containment.

Insert 1

In order for secondary containment integrity to be met, the secondary containment must function properly in conjunction with the operation of the Standby Gas Treatment 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.

With the reactor in the Run Mode, the Startup Mode, or the Hot Shutdown condition, if Secondary Containment Integrity is not maintained, Secondary Containment Integrity must be restored within 4 hours. The 4 hours provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during the Run Mode, the Startup Mode, and the Hot Shutdown condition. This time period also ensures that the probability of an accident (requiring Secondary Containment Integrity) occurring during periods where Secondary Containment Integrity is not maintained, is minimal.

BASES: 3.7 (Cont'd)

recently

If Secondary Containment Integrity cannot be restored within the required time period, the plant must be brought to a mode or condition in which the LCO does not apply.

Movement of irradiated fuel assemblies ~~or the fuel cask~~ in the secondary containment, alteration of the Reactor Core, and operations with the potential for draining the reactor vessel can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Alteration of the Reactor Core and movement of irradiated fuel assemblies ~~and the fuel cask~~ must be immediately suspended if Secondary Containment Integrity is not maintained. Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend operations with the potential for draining the reactor vessel to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until operations with the potential for draining the reactor vessel are suspended.

recently

BASES: 3.7 (Cont'd)

The Standby Gas Treatment System (SGTS) 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. To insure that the standby gas treatment system will be effective in removing radioactive contaminates from the Reactor Building air, the system is tested periodically to meet the intent of ANSI N510-1975. Laboratory charcoal testing will be performed in accordance with ASTM D3803-1989, except, as allowed by GL 99-02, testing can be performed at 70% relative humidity for systems with humidity control. Both standby gas treatment fans are designed to automatically start upon containment isolation and to maintain the Reactor Building pressure to approximately a negative 0.15 inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 100% capacity. This substantiates the availability of the operable train and results in no added risk; thus, reactor operation or refueling operation can continue. If neither train is operable, the plant is brought to a condition where the system is not required.

When the reactor is in cold shutdown or refueling the drywell may be open and the Reactor Building becomes the only containment system. During cold shutdown the probability and consequences of a DBA LOCA are substantially reduced due to the pressure and temperature limitations in this mode. However, for other situations under which significant radioactive release can be postulated, such as during operations with a potential for draining the reactor vessel, during core alterations, or during movement of recently irradiated fuel in the secondary containment, operability of standby gas treatment is required.

Both trains of the Standby Gas Treatment System are normally operable when secondary containment integrity is required. However, Specification 3.7.B.3 provides Limiting Conditions for Operation when one train of the Standby Gas Treatment System is inoperable. Provisional, continued operation is permitted since the remaining operable train is adequate to perform the required radioactivity release control function. If the applicable conditions of Specification 3.7.B.3 cannot be met, the plant must be placed in a mode or condition where the Limiting Conditions for Operation do not apply.

Entry into a refueling condition with one train of SBGTS inoperable is acceptable and there is no prohibition on mode or condition entry in this situation. In this case, the requirements of TS 3.7.B.3.b are sufficient to ensure that adequate controls are in place. During refueling conditions, accident risk is significantly reduced, and the primary activities of concern involve core alterations, movement of recently irradiated fuel assemblies~~,~~ and OPDRVs.

During refueling and cold shutdown conditions Specification 3.7.B.3.b provides for the indefinite continuance of refueling operations with one train of the Standby Gas Treatment System inoperable. When the seven-day completion time associated with Specification 3.7.B.3.b is not met and secondary containment integrity is required, the operable train of the Standby Gas Treatment System should immediately be placed into operation. This action ensures that the remaining train is operable, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. An alternative to placing the operable train of Standby Gas Treatment in operation is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Insert 1

"Recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within the previous 13 days, i.e. reactor fuel that has decayed less than 13 days following reactor shutdown. This minimum decay period is enforced to maintain the validity of the Fuel Handling Accident dose consequence analysis.