**VERMONT YANKEE**
**NUCLEAR POWER CORPORATION**

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BVY 01-85

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

**Subject:** **Vermont Yankee Nuclear Power Station**
**License No. DPR-28 (Docket No. 50-271)**
**Technical Specification Proposed Change No. 251**
**Removal of Primary Containment Isolation Valve Table, Revised SBGT Heater Rating and Miscellaneous Administrative Changes**

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications (TS). The proposed change consists of removal of TS Table 4.7.2, "Primary Containment Isolation Valves," and corresponding changes to support removal of this table. Additionally, a change to surveillance requirement TS 4.7.B.1.b, Standby Gas Treatment System inlet heater power rating, and other administrative changes are proposed.

Attachment 1 to this letter contains supporting information and the safety assessment of the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the marked-up version of the current Technical Specification and Bases pages showing the change requested. Attachment 4 contains the retyped Technical Specification and Bases pages.

VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

VY also believes that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued by May, 2002 for implementation within 90 days of its effective date. This will allow us to complete implementation of the amendment prior our Fall outage when resources are more limited.

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If you have any questions concerning this transmittal, please contact Mr. Jeffrey Meyer at (802) 258-4105.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Michael A. Balduzzi
Senior Vice President and Chief Nuclear Officer

STATE OF VERMONT )
)ss
WINDHAM COUNTY )

Then personally appeared before me, Michael A. Balduzzi, who, being duly sworn, did state that he is Senior Vice President and Chief Nuclear Officer of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2003

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS
Vermont Department of Public Service

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 251

Removal of Primary Containment Isolation Valve Table,
Revision of SBGT Heater Rating and Miscellaneous Administrative Changes

Supporting Information and Safety Assessment of Proposed Change

## DESCRIPTION OF CHANGES

Vermont Yankee (VY) proposes to change the Technical Specifications (TS) to remove Table 4.7.2, "Primary Containment Isolation Valves," and references to the table, from TS. Information from the TS table will be relocated to the Technical Requirements Manual (TRM), a licensee-controlled document, to which changes are adequately controlled and subject to 10CFR50.59. This change is consistent with Generic Letter 91-08, "Removal of Component Lists from Technical Specifications." In addition, TS surveillance requirement 4.7.B.1.b, and related TS Bases, are being changed to reflect the power input of the Standby Gas Treatment (SBGT) system duct inlet heater needed to meet the filter influent relative humidity design basis. Several other administrative changes are also proposed as described below.

### Proposed Changes

Table of Contents, page iii. Add section 3.7.E, Reactor Building Automatic Ventilation System Isolation Valves.

Page 75, Section 3.2 Protective Instrumentation Bases. Delete reference to "Note 1" in the second paragraph and delete text of Note 1 at bottom of page.

Page 101, Sections 3.5.A.4.a and b. Remove wording for a one-time 30 day Limiting Condition for Operation (LCO) during the 1989/1990 operating cycle.

Page 152, Surveillance Requirement 4.7.B.1.b. Revise the surveillance requirement from "9 kW" to "7.1 kW."

Page 158, Surveillance Requirement 4.7.D.1.a. Delete reference to Table 4.7.2 by changing the surveillance requirement to "The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure time at least once per operating cycle."

Pages 159 through 162, Table 4.7.2, Primary Containment Isolation Valves. Delete Table 4.7.2 in its entirety, including associated Notes. The following statement will be inserted on page 159 through 162: "Intentionally Blank."

Page 166a, Section 3.7.Bases. Revise description of 9 kW SBGT heater to 7.1 kW heater for consistency with revised surveillance specification 4.7.B.1.b.

Page 253, Section 5.3. Generalize specific referenced FSAR sections for reactor vessel description to Section 4 of the FSAR.

Page 257, Section 6.4. Remove reference to "in-plant" implementation of procedures to implement the ODCM.

## REASON/BASIS FOR CHANGES

The Table of Contents is changed to add a new section 3.7.E. This new section was previously added by License Amendment 197, however, the Table of Contents was not revised at that time and currently does not list this section. The change is administrative and is being proposed to provide consistency between the Table of Contents and the Specifications.

Section 3.5.A.4.a and b has provisions for a one-time LPCI subsystem 30-day LCO for the 1989/1990 operating cycle. This wording is obsolete, no longer required and is proposed to be eliminated. This is an administrative change to delete obsolete information that is no longer applicable.

VY TS Table 4.7.2 contains a list of primary containment isolation valves. Generic Letter 91-08 provides guidance to remove component lists, such as containment isolation valves, from the TS as an acceptable alternative to identifying numerous individual components. It also allows changes to component lists without the need to process license amendments. Such changes are acceptable provided that they do not alter existing TS requirements or the types of components to which the requirements apply. With the proposed change to relocate Table 4.7.2, TS Section 3.7.D, "Primary Containment Isolation Valves," will provide a general description of the components to which the TS requirement applies. In addition, requirements related to operability, applicability, and surveillance, including the performance of testing to ensure operability, will be retained in TS under the proposed changes. The removal of Table 4.7.2 from the VY TS does not alter existing requirements or those components to which they apply. The table is being relocated to the VY TRM, a licensee controlled document, with changes subject to 10CFR50.59.

Section 3.2, page 75 refers to Table 4.7.2 and it is proposed to delete reference to the TS Table. This is an administrative change and is proposed to provide consistency with removal of the table from TS.

TS surveillance 4.7.B.1.b specifies a SBGT inlet heater power input of at least 9 kW and is based on the original, nominal rating of the heater. The VY design basis for SBGT train inlet heater capacity is to raise the temperature of the influent flow stream, thereby reducing the relative humidity to the charcoal beds in order to minimize the introduction of moisture which can adversely affect charcoal adsorption efficiency. The inlet power needed to achieve the design assumption of 70% relative humidity is calculated to be 7.1 kW. Therefore, VY proposes to revise the TS surveillance and associated Bases to require inlet heater power input of at least 7.1 kW. Additionally, the Bases are revised to identify this as the 7.1 kW heater for consistency.

Section 5.3, Reactor Vessel Design Features, is being revised because the listed, specific subsection/table FSAR references are not correct. The new reference is to the general section of the FSAR where the reactor vessel design information is located. This is an administrative change.

Section 6.4, Procedures, is being revised to generalize the reference to procedures that implement the ODCM. Currently, a more narrow grouping of these procedures could be interpreted to apply since the words "in-plant" implementation are listed. The reference to "in-plant" will be removed. This is an administrative change.

## SAFETY ASSESSMENT

Revision of the Table of Contents to include a new section, approved by a previous license amendment, is an administrative change and has no impact on plant safety.

The proposed change to relocate TS Table 4.7.2 does not affect the ability of the primary containment isolation system to perform its safety function. TS requirements for primary containment isolation valve operability and surveillance are not reduced, nor does the change alter the components to which the requirements apply. Information contained in Table 4.7.2 will be relocated to the TRM and referenced, as appropriate, by applicable plant procedures. Associated references to the relocated Table in other TS sections are proposed to be deleted for consistency. Primary containment isolation valve operability and surveillance will continue to be maintained in a manner consistent with assumptions contained in the VY accident and transient analyses. Therefore this change has no impact on plant safety.

The proposed change to the SBGT inlet heater input power surveillance and Bases more correctly specifies the minimum capability of the heater to achieve the design basis of 70% relative humidity for the system influent. There are no changes to system components or operation of the system. The VY accident analysis assumes that SBGT operates in accordance with its design basis to remove methyl iodide from the plant airborne effluent, thereby meeting 10CFR100 limits. SBGT charcoal bed removal efficiency is affected by the presence of moisture in the flow stream, and the design basis value for limiting moisture is no more than 70% relative humidity. The inlet heater must demonstrate sufficient input power to raise flow stream temperature to lower relative humidity. A SBGT inlet heater input power of 7.1 kW will result in charcoal bed inlet relative humidity of no greater than 70% for design basis accident conditions. Therefore this change has no impact on plant safety.

Removal of an obsolete LPCI system 30-day LCO that was only applicable for the 1989-1990 operating cycle is an administrative change, has no impact on plant systems, structures or components and therefore has no affect on plant safety.

Generalizing the FSAR sections referenced in Section 5.3, and removing the words “in-plant” in Section 6.4.I, are administrative changes, have no impact on plant systems, structures or components and therefore have no affect on plant safety.

<u>Summary</u>

In summary, the removal of TS Table 4.7.2 permits administrative control of changes to the list of primary containment isolation valves and is proposed in accordance with NRC guidance contained in Generic Letter 91-08. Revision of the SBGT inlet heater input power surveillance requirement from 9 kW to 7.1 kW does not adversely affect the ability of the system to meet 10CFR100 limits. The proposed change does not significantly reduce the margin of safety for the containment isolation function or SBGT system function. The other changes are administrative in nature and do not affect any system operation, function or safety analyses.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 251

Removal of Primary Containment Isolation Valve Table,
Revision of SBGT Heater Rating and Miscellaneous Administrative Changes

Determination of No Significant Hazards Consideration

# Determination of No Significant Hazards Consideration

Description of Amendment Request:

Vermont Yankee (VY) has determined that the proposed changes to Technical Specifications (TS), which removes Table 4.7.2 (and associated references to the Table) in accordance with guidance provided in Generic Letter 91-08, revises the minimum Standby Gas Treatment System (SBGT) inlet heater surveillance power input from 9 kW to 7.1 kW, eliminates an obsolete 30-day Limiting Condition for Operation (LCO), revises the Table of Contents, generalizes a reference to specific FSAR sections and generalizes the TS implementing procedures for the Off-Site Dose Calculation Manual (ODCM) listed in section 6.4, do not involve a Significant Hazards Consideration. In support of this determination, each of the three (3) standards set forth in 10CFR50.92 is provided below.

Basis for No Significant Hazards Determination:

Pursuant to 10CFR50.92, VY has reviewed the proposed changes and concluded that they do not involve a significant hazards consideration since the proposed changes satisfy the criteria in 10CFR50.92(c).

1. The operation of the Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes consist of removal of the primary containment isolation valve component list from the VY TS, revision of the SBGT inlet heater surveillance minimum power rating and other administrative changes. The probability of occurrence of a previously evaluated accident is not increased because neither containment isolation nor the SBGT heater are accident initiators, and the proposed changes do not impact any accident initiating conditions. The consequences of an accident previously evaluated are not increased because the proposed changes do not impact the ability of containment to restrict, or SBGT to filter, the release of any fission product radioactivity to the environment. The proposed changes to remove the primary containment isolation valve component list from TS, relocate the information to a licensee controlled document, and to change the SBGT inlet heater power input surveillance requirement, will have no significant impact on any safety related structures, systems or components. The TS requirements for the primary containment isolation valves and SBGT operability and surveillance will not be changed. Additionally, the administrative changes do not affect any system operation or function.

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

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes do not involve any physical alteration of plant equipment and do not change the method by which any safety-related system performs its function. No new or different types of equipment will be installed. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

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

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   The proposed administrative changes, the removal of the primary containment isolation valve component list from TS and the change to the SBGT inlet heater power input surveillance requirement, do not alter the TS requirements for containment integrity, containment isolation, SBGT operability, or adversely affect their capability. The changes will not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

   The proposed changes do not impact design margins of the primary containment isolation system, SBGT or any other system to perform their safety functions. The essential safety functions of providing primary containment integrity and providing filtration of airborne radioactive releases, are maintained. There is no physical or operational change being made which would alter the sequence of events, plant response, or margins in existing safety analyses. The proposed changes result in no impact on analyzed accident event precursors or effects.

   These proposed changes do not alter the physical design of the plant. There is no change in methods of operation. The proposed changes do not alter the means by which primary containment isolation capability is maintained and SBGT is operated.

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

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 251

Removal of Primary Containment Isolation Valve Table,
Revision of SBGT Heater Rating and Miscellaneous Administrative Changes

Marked-up Version of the Current Technical Specifications

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BASES:

## 3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves ~~(Note 1)~~ are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" $H_2O$ indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated.

~~Note 1 - Isolation valves are grouped as listed in Table 4.7.2.~~

## 3.5 LIMITING CONDITION FOR OPERATION

4. a. ~~From and after the date that a LPCI Subsystem is made or found to be inoperable due to failure of the associated UPS, reactor operation is permissible only during the succeeding thirty days, for the 1989/90 operating cycle, unless it is sooner made operable, provided that during that time the associated motor control center (89A or 89B) is powered from its respective maintenance tie, all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the emergency diesel generators shall be operable, the requirements of Specification 3.10.A.4 are met, and the 4160 volt tie line to the Vernon Hydro is the operable delayed access power source.~~

~~b.~~ From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, ~~other than failure of the UPS during the 1989/90 operating cycle, or Specification 3.5.A.4.a is not met,~~ reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided

## 4.5 SURVEILLANCE REQUIREMENT

4. When a LPCI Subsystem is made or found to be inoperable, the active components of the redundant LPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours (except the Recirculation System discharge valves).

3.5 LIMITING CONDITION FOR OPERATION

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that during that time all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

5. All recirculation pump discharge valves and bypass valves shall be operable or closed prior to reactor startup.

6. If the requirements of Specifications 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. <u>Containment Spray Cooling Capability</u>

1. Both containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a Containment Cooling Subsystem may be inoperable for thirty days.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

5. Recirculation pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds and bypass valves shall be tested for operability in accordance with Specification 4.6.E.

B. <u>Containment Spray Cooling Capability</u>

1. Surveillance of the drywell spray loops shall be performed as follows. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

2. When a Containment Cooling Subsystem is made or found to be inoperable, the active components of the redundant Containment Cooling Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

## 3.7 LIMITING CONDITIONS FOR OPERATION

$\Delta P$ is reduced to <1.7) during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-suppression chamber vacuum breakers, and the suppression chamber-reactor building vacuum breakers, and SGTS testing.

d. If the specifications of 3.7.A.9.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

B. Standby Gas Treatment System

1. a. Except as specified in Specification 3.7.B.3.a below, whenever the reactor is in Run Mode or Startup Mode or Hot Shutdown condition, both trains of the Standby Gas Treatment System shall be operable at all times when secondary containment integrity is required.

   b. Except as specified in Specification 3.7.B.3.b below, whenever the reactor is in Refuel Mode or Cold Shutdown condition, both trains of the Standby Gas

## 4.7 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

1. At least once per operating cycle, not to exceed 18 months, the following conditions shall be demonstrated.

   a. Pressure drop across the combined HEPA and charcoal filter banks is less than 6 inches of water at 1500 cfm ±10%.

   b. Inlet heater input is at least ~~8~~ kW.

7.1

## 3.7 LIMITING CONDITIONS FOR OPERATION

D. <u>Primary Containment Isolation Valves</u>

1. During reactor power operating conditions all containment isolation valves and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

2. In the event any containment isolation valve becomes inoperable, reactor power operation may continue provided at least one containment isolation valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

## 4.7 SURVEILLANCE REQUIREMENTS

D. <u>Primary Containment Isolation Valves</u>

1. Surveillance of the primary containment isolation valves should be performed as follows:

   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and ~~the closure times specified in Table 4.7.2~~ at least once per operating cycle.

   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

   c. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

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TABLE 4.7.2

PRIMARY CONTAINMENT ISOLATION VALVES

| Isolation Group (1) | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 1 | Main Steam Line Isolation (2-80A-D & 2-86A-D) | 4 | 4 | 5 (Note 2) | Open | GC |
| 1 | Main Steam Line Drain (2-74, 2-77) | 1 | 1 | 35 | Closed | SC |
| 1 | Recirculation Loop Sample Line (2-39, 2-40) | 1 | 1 | 5 | Closed | SC |
| 2 | RHR Discharge to Radwaste (10-57, 10-66) | | 2 | 25 | Closed | SC |
| 2 | Drywell Floor Drain (20-82, 20-83) | | 2 | 20 | Open | GC |
| 2 | Drywell Equipment Drain (20-94, 20-95) | | 2 | 20 | Open | GC |
| 3 | Drywell Air Purge Inlet (16-19-9) | | 1 | 10 | Closed | SC |
| 3 | Drywell Air Purge Inlet (16-19-8) | | 1 | 10 | Closed | SC |
| 3 | Drywell Purge & Vent Outlet (16-19-7A) | | 1 | 10 | Closed* | SC |
| 3 | Drywell Purge & Vent Outlet Bypass (16-19-6A) | | 1 | 10 | Closed | SC |
| 3 | Drywell & Suppression Chamber Main Exhaust (16-19-7) | | 1 | 10 | Closed* | SC |
| 3 | Suppression Chamber Purge Supply (16-19-10) | | 1 | 10 | Closed | SC |
| 3 | Suppression Chamber Purge & Vent Outlet (16-19-7B) | | 1 | 10 | Closed | SC |
| 3 | Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B) | | 1 | 10 | Open | GC |

* Valves 16-19-7 and 16-19-7A shall have stops installed to limit valve opening to 50° or less.

TABLE 4.7.2
(Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

| Isolation Group (1) | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 3 | Exhaust to Standby Gas Treatment System (16-19-6) | | 1 | 10 | Open | GC |
| 3 | Containment Purge Supply (16-19-23) | | 1 | 10 | Closed | SC |
| 3 | Containment Makeup Supply (16-20-22A) | | 1 | NA | Closed | SC |
| 3 | Containment Makeup Supply (16-20-20, 16-20-22B) | | 2 | 5 | Open | GC |
| 5 | Reactor Cleanup System (12-15, 12-18) | 1 | 1 | 25 | Open | GC |
| 6 | HPCI (23-15, 23-16) | 1 | 1 | 55 | Open | GC |
| 6 | RCIC (13-15, 13-16) | 1 | 1 | 20 | Open | GC |
| | Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B) | | 2 | NA | Closed | SC |
| | Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B) | | 2 | NA | Closed | Process |
| 3 | Containment Air Sampling (VG 23, VG 26, 109-76A&B) | | 4 | 5 | Open | GC |
| | Feedwater Check Valves (V2-27A, -96A, -28A, -28B) | | | NA | Open | Process |

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TABLE 4.7.2
(Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

| Isolation Group (1) | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 2 | RHR Return to Suppression Pool (10-39A, B) | | 2 | 70 | Closed | SC |
| 2 | RHR Return to Suppression Pool (10-34A, B) | | 2 | 120 | Closed | SC |
| 2 | RHR Drywell Spray (10-26A, B & 10-31A, B) | | 4 | 70 | Closed | SC |
| 2 | RHR Suppression Chamber Spray (10-38A, B) | | 2 | 45 | Closed | SC |
| 3 | Containment Air Compressor Suction (72-38A, B) | | 2 | 20 | Open | GC |
| 4 | RHR Shutdown Cooling Supply (10-18, 10-17) | 1 | 1 | 28 | Closed | SC |
| | Standby Liquid Control Check Valves (11-16, 11-17) | 1 | 1 | NA | Closed | Proc. |
| * | Hydrogen Monitoring (109-75 A, 1-4; 109-75 B-D, 1-2) Sampling Valves - Inlet | | 10 | NA | NA | NA |
| * | Hydrogen Monitoring (VG-24, 25, 33, 34) | | 4 | NA | NA | NA |

* These valves are remote manual sampling valves which do not receive an isolation signal. Only one valve in each line is required to be operable.

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TABLE 4.7.2 NOTES

1. Isolation signals are as follows:

Group 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Low-low reactor water level
2. High main steam line radiation
3. High main steam line flow
4. High main steam line tunnel temperature
5. Low main steam line pressure (run mode only)
6. Condenser low vacuum

Group 2: The valves in Group 2 are closed upon any one of the following conditions*:

1. Low reactor water level
2. High drywell pressure

Group 3: The valves in Group 3 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High/low radiation - reactor building ventilation exhaust plenum or refueling floor

Group 4: The valves in Group 4 are closed upon any one of the following conditions:

1. Low reactor water level
2. High drywell pressure
3. High reactor pressure

Group 5: The valves in Group 5 are closed upon low reactor water level.

Group 6: The valves in Group 6 are closed upon any signal representing a steam line break in the HPCI system's or RCIC system's respective steam line. The signals indicating a steam line break for the respective steam line are as follows:

1. High steam line space temperature
2. High steam line flow
3. Low steam line pressure
4. High temperature in the main steam line tunnel (30 minute delay for the HPCI and the RCIC)

2. The closure time shall not be less than 3 seconds.

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* Valves V10-39A/B, V10-34A/B, V10-26A/B, V10-31A/B and V10-38A/B are closed upon either 1) low-low reactor water level and low reactor pressure or 2) high drywell pressure.

BASES: 3.7 (Cont'd)

An alternate electrical power source for the purposes of Specification 3.7.B.1.b shall consist of either an Emergency Diesel Generator (EDG) or the Vernon Hydro tie line. Maintaining availability of the Vernon Hydro tie line as an alternative to one of the EDGs in this condition provides assurance that standby gas treatment can, if required, be operated without placing undue constraints on EDG maintenance availability. Inoperability of both trains of the SGTS or both EDGs during refueling operations requires suspension of activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Use of the SGTS, without the fan and the ~~9~~ 7.1 kW heater in operation, as a vent path during torus venting does not impact subsequent adsorber capability because of the very low flows and because humidity control is maintained by the standby 1 kW heaters, therefore operation in this manner does not accrue as operating time.

D. Primary Containment Isolation Valves

Generally, double isolation valves are provided on lines that penetrate the primary containment and communicate directly with the reactor vessel and on lines that penetrate the primary containment and communicate with the primary containment free space. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

E. Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs)

The function of the RBAVSIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (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. The valves covered by this Limiting Condition for Operation are included in the Inservice Testing Program.

In the event that there are one or more RBAVSIVs inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. The required action must be completed within the eight hour or four hour completion time, as applicable. The specified time periods are reasonable considering the time required to isolate the penetration, and the probability of a DBA occurring during this short time.

If any required action or completion time cannot be met as a result of one or more inoperable RBAVSIVs, the plant must be placed in a mode or condition where the Limiting Condition for Operation does not apply. To achieve this status during reactor power operation, the reactor must be brought to at least hot shutdown within 12 hours and to cold shutdown within 36 hours. If applicable, core alterations and the movement of irradiated fuel assemblies and the fuel cask in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

# 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 the Vermont Yankee Nuclear Power Corporation 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.

No part of the site shall be sold or leased and no structure shall be located on the site except structures owned by the Vermont Yankee Nuclear Power Corporation or related utility companies and used in conjunction with normal utility operations.

## 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 ~~shall be as described in Table 4.2-3 of the FSAR. The applicable design codes shall be as described in subsection 4.2 of the FSAR.~~

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.

## 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.

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

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.

6.2 ORGANIZATION (Cont'd)

C. Unit Staff Qualifications

Each member of the unit staff shall meet or exceed the minimum qualifications of the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants," except for the radiation protection manager who shall meet the qualifications of Regulatory Guide 1.8, Revision 1 (September 1975) and the Shift Engineer, who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.3 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

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 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.

F. Surveillance and testing requirements.

G. Fire protection program implementation.

H. Process Control Program in-plant implementation.

I. Off-Site Dose Calculation Manual ~~in-plant~~ 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:

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 251

Removal of Primary Containment Isolation Valve Table,
Revision of SBGT Heater Rating and Miscellaneous Administrative Changes

Retyped Technical Specification Pages

TABLE OF CONTENTS
(Continued)

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" $H_2O$ indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated.

3.5 LIMITING CONDITION FOR OPERATION

4. From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during that time all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

4.5 SURVEILLANCE REQUIREMENT

4. When a LPCI Subsystem is made or found to be inoperable, the active components of the redundant LPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours (except the Recirculation System discharge valves).

## 3.5 LIMITING CONDITION FOR OPERATION

5. All recirculation pump discharge valves and bypass valves shall be operable or closed prior to reactor startup.

6. If the requirements of Specifications 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. Containment Spray Cooling Capability

1. Both containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a Containment Cooling Subsystem may be inoperable for thirty days.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

## 4.5 SURVEILLANCE REQUIREMENT

5. Recirculation pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds and bypass valves shall be tested for operability in accordance with Specification 4.6.E.

B. Containment Spray Cooling Capability

1. Surveillance of the drywell spray loops shall be performed as follows. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

2. When a Containment Cooling Subsystem is made or found to be inoperable, the active components of the redundant Containment Cooling Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.7 LIMITING CONDITIONS FOR OPERATION

ΔP is reduced to <1.7) during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-suppression chamber vacuum breakers, and the suppression chamber-reactor building vacuum breakers, and SGTS testing.

d. If the specifications of 3.7.A.9.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

B. Standby Gas Treatment System

1. a. Except as specified in Specification 3.7.B.3.a below, whenever the reactor is in Run Mode or Startup Mode or Hot Shutdown condition, both trains of the Standby Gas Treatment System shall be operable at all times when secondary containment integrity is required.

b. Except as specified in Specification 3.7.B.3.b below, whenever the reactor is in Refuel Mode or Cold Shutdown condition, both trains of the Standby Gas

4.7 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

1. At least once per operating cycle, not to exceed 18 months, the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA and charcoal filter banks is less than 6 inches of water at 1500 cfm ±10%.

b. Inlet heater input is at least 7.1 kW.

## 3.7 LIMITING CONDITIONS FOR OPERATION

### D. Primary Containment Isolation Valves

1. During reactor power operating conditions all containment isolation valves and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

2. In the event any containment isolation valve becomes inoperable, reactor power operation may continue provided at least one containment isolation valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

## 4.7 SURVEILLANCE REQUIREMENTS

### D. Primary Containment Isolation Valves

1. Surveillance of the primary containment isolation valves should be performed as follows:

   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure time at least once per operating cycle.

   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

   c. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

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BASES: 3.7 (Cont'd)

An alternate electrical power source for the purposes of Specification 3.7.B.1.b shall consist of either an Emergency Diesel Generator (EDG) or the Vernon Hydro tie line. Maintaining availability of the Vernon Hydro tie line as an alternative to one of the EDGs in this condition provides assurance that standby gas treatment can, if required, be operated without placing undue constraints on EDG maintenance availability. Inoperability of both trains of the SGTS or both EDGs during refueling operations requires suspension of activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk.

Use of the SGTS, without the fan and the 7.1 kW heater in operation, as a vent path during torus venting does not impact subsequent adsorber capability because of the very low flows and because humidity control is maintained by the standby 1 kW heaters, therefore operation in this manner does not accrue as operating time.

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