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December 6, 2004

Docket No. 50-271
BVY 04-118

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 270
Administrative Changes

Dear Sir:

Pursuant to 10CFR50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby proposes to amend the Facility Operating License, DPR-28, for Vermont Yankee Nuclear Power Station (VY) by incorporating the attached proposed change into the VY Technical Specifications (TS). The proposed changes are administrative in nature and do not materially change any technical requirements.

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

In accordance with 10CFR50.91, a copy of this application and the associated attachments are being submitted to the designated Vermont State official.

There are no new commitments being made in this submittal.

VY requests approval of the proposed amendment by October 8, 2005 with the amendment being implemented within 30 days. If you have any questions or require additional information, please contact Mr. James M. DeVincentis at (802) 258-4236.

A001

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

Executed on December 6, 2004.

Sincerely,

Jay K. Thayer
Site Vice President
Vermont Yankee Nuclear Power Station

Attachments (3)

cc: Mr. Richard B. Ennis, Project Manager
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U.S. Nuclear Regulatory Commission
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Vermont Yankee Nuclear Power Station
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ATTACHMENT 1 TO BVY 04-118

Vermont Yankee Nuclear Power Station
Technical Specification Proposed Change No. 270
Administrative Changes

DESCRIPTION, SAFETY ASSESSMENT, AND DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

ENTERGY NUCLEAR OPERATIONS, INC.
VERMONT YANKEE NUCLEAR POWER STATION
DOCKET NO. 50-271

1. PROPOSED CHANGES

Entergy Nuclear Operations, Inc. (Entergy) proposes to revise the Vermont Yankee Nuclear Power Station (VY) Technical Specifications (TS) by correcting references as a result of changes to 10CFR and to previous License Amendments, deleting obsolete or redundant TS requirements and surveillances. Renumbering of the pages and specifications is identified as a result of the proposed changes as well as corresponding BASES changes. As such, these changes are administrative in nature and do not materially change any technical requirement.

The below Table details each proposed change (as identified by the [number] on the Marked-up Technical Specification pages included as Attachment 2) and provides the basis and safety assessment for each change.

Change #	Current License Requirement	Proposed Change
1	VY Operating License Condition 3.D 'Records' currently states "Entergy Nuclear Operations, Inc. shall keep facility operating records in accordance with the requirements of the Technical Specifications."	It is proposed that Operating License Condition 3.D "Records" be deleted in its entirety. In its place will be the phrase "This paragraph deleted by Amendment No. (xxx), (month day, year)." The blanks will be completed by Entergy pending approval of the proposed change, as communicated by the NRC and provided to the NRC just prior to issuance.
Basis/Safety Assessment: In License Amendment # 171, Administrative Section 6.6 "Plant Operating Records" was removed from the Technical Specifications and relocated to the quality assurance manual. However, during this Amendment, VY failed to identify that License Condition 3.D referenced this section of the Technical Specifications. Accordingly, this proposed change proposes to remove this License Condition. This is an administrative change since the requirements to maintain records are specifically called out in other regulatory documents. Specifically, the requirement for retention of records related to activities affecting quality is contained in 10CFR50, Appendix B, Criterion XVII and other sections of 10CFR50 (e.g.; 10 CFR 50.71) that are applicable to VY. As such, the requirement to maintain records in accordance with the quality assurance manual does not need to be duplicated in the License. It is noted that this change does not propose to revise any of the record keeping requirements currently contained within the quality assurance manual.		

Change #	Current Technical Specifications	Proposed Change
2	Technical Specification (TS) 4.7.C.1.a and 4.7.C.1.b discuss the details of secondary containment testing that is required to be conducted during VY's preoperational phase or during the first operating cycle.	It is proposed to delete TS 4.7.C.1.a and 4.7.C.1.b since they no longer apply. Accordingly, TS 4.7.C.1.c would be renumbered to become 4.7.C.1.
Basis/Safety Assessment: The requirement to perform preoperational tests and tests during the first operating cycle is deleted since preoperational testing and testing during the first operating cycle has already been completed. As such, this is an administrative change since it does not delete any applicable surveillance requirements.		

Change #	Current Technical Specifications	Proposed Change
3	TS 4.7.C.1.c requires, in part, that secondary containment be tested at each refueling outage prior to refueling.	It is proposed to remove the portion of the requirement contained within TS 4.7.C.1.c (which is to be renumbered as TS 4.7.C.1 per Change # 2 above) to perform a special secondary containment test during each refueling outage prior to refueling.
Basis/Safety Assessment: Technical Specification Surveillance Requirement 4.7.C.1.c (SR 4.7.C.1.c), which is to be renumbered as SR 4.7.C.1 per Change # 2 above, requires the performance of the secondary containment capability test with the Standby Gas Treatment subsystem "prior to refueling." It is proposed to delete the requirement to perform this test "prior to refueling." TS Surveillances must be continually met (per TS SR 4.0.1[1]), thus if SR 4.7.C.1 is not met the appropriate actions of TS 3.7.C.2, 3 or 4 shall be taken. While conducting this test prior to refueling is a good plant practice, it is not necessary to ensure the OPERABILITY of the secondary containment provided the Technical Specification Surveillance Requirement is current. Removal of this redundant surveillance test is consistent with STS surveillance requirements in section 3.6.4.3, Standby Gas Treatment System.		

[1] It is noted that SR 4.0.1 has been requested in Technical Specification Proposed Change # 261 and is awaiting review and approval. Reference VY letter to USNRC, BVY 03-57, "Missed Surveillances and Adoption of a Technical Specification Bases Control Program using the Consolidated Line Item Improvement Process," dated September 16, 2003.

Change #	Current Technical Specifications	Proposed Change
4	Technical Specification Surveillance Requirement (SR) 4.10.B.4 requires that the requirements of SR 4.5.A.4 be satisfied when one Uninterruptible Power System or its associated Motor Control Center is inoperable. SR 4.5.A.4 was deleted as a part of License Amendment # 209.	It is proposed to delete SR 4.10.B.4.
Basis/Safety Assessment: Technical Specification Surveillance Requirement (SR) 4.10.B.4 requires that the requirements of SR 4.5.A.4 be satisfied when one Uninterruptible Power System or its associated Motor Control Center is inoperable. SR 4.5.A.4 was deleted as a part of License Amendment # 209[2]. Since SR 4.10.B.4 requires the performance of an additional SR which has been deleted, it is considered an administrative change. In addition, TS 3.10.B.4 "480 V Uninterruptible Power Systems" requires that the Limiting Conditions for Operation of specification 3.5.A.4 be satisfied if one Uninterruptible Power System or its associated Motor Control Center is inoperable. Accordingly, the requirements for system operability and the corresponding surveillances are covered as part of the TS 3.5.A and 4.5.A. Since the proposed change will not delete or remove any required surveillances, and since the requirements for system operability are unaffected, the proposed change is considered administrative.		

Change #	Current Technical Specifications	Proposed Change
5	Technical Specification (TS) 6.2.A.1 reads in part: "...These requirements [for the Onsite and Offsite Organizations] shall be documented in the Vermont Yankee Operational Quality Assurance Manual."	It is proposed that TS 6.2.A.1 be revised to recognize that Vermont Yankee has adopted the Quality Assurance Program Manual.
Basis/Safety Assessment: The proposed changes to TS 6.2.A.1 will only revise the title of the VY quality assurance program manual. None of the lines of authority, responsibility, qualifications, etc. shall be revised for the Onsite and Offsite Organizations by this proposed change. Accordingly, this change is administrative only.		

2 Reference USNRC letter to VY, NVY 02-71, "Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: Operability of Alternate Trains (TAC No. MB2760)," dated August 14, 2002.

Change #	Current Technical Specifications	Proposed Change
6	Technical Specification (TS) 4.6.E.2 currently states "Operability testing of safety-related pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except..."	The proposed change would require the testing of safety related pumps and valves be performed in accordance with "the Code of Record as required by 10 CFR 50.55a, except..." Similar wording changes are being proposed to various BASES pages to remove the out of date reference to ASME Section XI.

Basis/Safety Assessment:

Technical Specification (TS) 4.6.E.2 currently requires Operability testing of safety related pumps and valves, also known as Inservice Testing (IST), to be performed in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f). The IST requirements were once part of ASME Section XI; however, this has since been relocated to a different ASME document. Specifically, the Code and Addenda currently required by 10 CFR 50.55a is actually the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). Accordingly, the proposed change would delete an incorrect reference and instead reference the Code of Record as required by 10 CFR 50, Section 50.55a. This change would not modify VY's IST program or the testing performed on safety related equipment, accordingly, the proposed change is considered administrative.

It is noted that similar wording changes are being proposed to various BASES pages to remove the out of date reference to ASME Section XI. These changes are proposed for pages 98, 110, 143 and 223.

Change #	Current Technical Specifications	Proposed Change
7	The second paragraph of Technical Specification (TS) 4.6.E.1 reads: "Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter or in accordance with alternate measures approved by NRC Staff." (emphasis added)	No change is being proposed to this page at this time. The word "alternte" was inadvertently corrected to "alternate" without markup or justification during License Amendment (LA) #219.
Basis/Safety Assessment: No change is being proposed to this page at this time. However, justification for a change that was inadvertently made as a part of LA #219[3] is being provided. The word "alternte" was inadvertently corrected to "alternate" without markup or justification. The change which occurred as part of LA #219 simply corrected a typographical error. However, since this change was not accounted for during that License Amendment, it is being provided as part of this change request. It is proposed that once approved by the Staff, this page will have a new license amendment number, but the page will be void of any rev bars since no changes are being proposed. Since the change that occurred during LA 219 was the correction of a typographical error, and since this change only serves to document and receive acknowledgement of the correction, this change is considered administrative.		

[3] Reference USNRC letter to VY, NVY 04-031 "Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: Implementation of ARTS /MELLLA (TAC No. MB8070)," dated April 14, 2004.

Change #	Current Technical Specifications	Proposed Change
8	Technical Specification (TS) 3.7.C.5 reads: "Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F." The corresponding SR 4.7.C.5 requires that the subject lower compartment openings shall be checked closed daily.	It is proposed that TS 3.7.C.5 and SR 4.7.C.5 be deleted in their entirety and relocated to FSAR.

Basis/Safety Assessment:

TS 3.7.C.5 and SR 4.7.C.5 provides requirements to close / check closed the Core Spray and LPCI lower compartment doors at all times except during passage or when the reactor coolant temperature is less than 212°F. It is proposed that these requirements be relocated to the FSAR. The Core spray and LPCI lower compartment doors are related to flood protection and are not necessary for ensuring the secondary containment is maintained OPERABLE.
TS 3.7.C requires the secondary containment to be OPERABLE. These requirements are adequate for ensuring the secondary containment is maintained OPERABLE. As such, the proposed relocated details are not required to be in the Technical Specifications to provide adequate protection of the public health and safety. Changes to the FSAR will be controlled by 10 CFR 50.59.

2. REGULATORY SAFETY ANALYSIS

2.1 No Significant Hazards Consideration

Entergy Nuclear Operation, Inc. (Entergy) proposes to revise the Vermont Yankee Nuclear Power Station (VY) Technical Specifications (TS) by correcting references as a result of changes to 10CFR and to previous License Amendments, deleting obsolete or redundant TS requirements and surveillances. Renumbering of the pages and specifications is identified as a result of the proposed changes as well as corresponding BASES changes. As such, these changes are administrative in nature and do not materially change any technical requirement.

Pursuant to 10CFR50.92, Entergy has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c).

1. The operation of 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 are administrative or editorial in nature and do not involve any physical changes to the plant. The changes do not revise the methods of plant operation which could increase the probability or consequences of accidents. No new modes of operation are introduced by the proposed changes such that a previously evaluated accident is more likely to occur or more adverse consequences would result.

   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.

   These changes are administrative or editorial in nature and do not affect the operation of any systems or equipment, nor do they involve any potential initiating events that would create any new or different kind of accident. There are no changes to the design assumptions, conditions, configuration of the facility, or manner in which the plant is operated and maintained. The changes do not affect assumptions contained in plant safety analyses or the physical design and/or modes of plant operation. Consequently, no new failure mode is introduced.

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

   There are no changes being made to the Technical Specification (TS) safety limits or safety system settings. The operating limits and functional capabilities of systems, structures and components are unchanged as a result of these administrative and editorial changes. These changes do not affect any equipment involved in potential initiating events or plant response to accidents. There is no change to the basis for any TS that is related to the establishment, or maintenance of, a nuclear safety margin.

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

2.2 Environmental Consideration

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental assessment needs to be prepared in connection with the proposed amendment.

ATTACHMENT 2 TO BVY 04-118

Vermont Yankee Nuclear Power Station
Technical Specification Proposed Change No. 270
Administrative Changes

MARKED-UP VERSION OF TECHNICAL SPECIFICATION PAGES

ENTERGY NUCLEAR OPERATIONS, INC.
VERMONT YANKEE NUCLEAR POWER STATION
DOCKET NO. 50-271

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

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

A. Maximum Power Level

Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1593 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment 208, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

C. Reports

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

D. ~~Records~~ THIS PARAGRAPH DELETED By Amendment No.___, (DATE)______

~~Entergy Nuclear Operations, Inc. shall keep facility operating records in accordance with the requirements of the Technical Specifications.~~ [1]

E. Environmental Conditions

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

BASES: 3.4 & 4.4 (Cont'd)

B. Operation With Inoperable Components

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with ~~ASME Section XI requirements~~ THE Requirements OF Specification 4.6.E. [6]

C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. Fixing the pump flow rate in Specification 3.4.C.3 at the minimum flow rate of 35 gpm conservatively establishes a system parameter that can be used in satisfying the ATWS requirement, as well as the original system design basis. If the product of the expression in Specification 3.4.C.3 is equal to or greater than unity, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4).

BASES:

3.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This Specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss-of-coolant analyses, the Core Spray and LPCI Systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit the accident-caused core conditions as specified in 10CFR50, Appendix K. The analyses consider appropriate combinations of the two Core Spray Subsystems and the two LPCI Subsystems associated with various break locations and equipment availability in accordance with required single failure assumptions. (Each LPCI Subsystem consists of the LPCI pumps, the recirculation pump discharge valve, and the LPCI injection valve which combine to inject torus water into a recirculation loop.)

The LPCI System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI and the Core Spray Systems provide adequate cooling for break areas up to and including the double-ended recirculation line break without assistance from the high pressure emergency Core Cooling Subsystems.

Specification 3.5.A.1 is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal with reactor pressure less than the RHR shutdown cooling permissive pressure, if capable of being manually realigned (remote) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during Hot Shutdown, if necessary.

The intent of these specifications is to prevent startup from the cold condition without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ~~ASME Section XI~~ THE requirements ~~referenced in~~ OF Specification 4.6.E. [6]

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B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR System is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference: Section 14.6.3.3.2 FSAR.

Each Containment Cooling Subsystem consists of two RHR service water pumps, 1 heat exchanger, and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger, and 1 RHR pump has sufficient capacity to perform the cooling function. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ~~ASME Section XI~~ THE requirements ~~referenced in~~ OF Specification 4.6.E. [6]

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3.6 LIMITING CONDITIONS FOR OPERATION

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 150 psig and temperature greater than 350°F, all safety valves and at least three of the four relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 150 psig and 350°F within 24 hours.

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

4.6 SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

   Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter ~~or in~~ accordance with alternate measures approv~~ed by NRC~~ Staff. [7]

3.6 LIMITING CONDITIONS FOR OPERATION

THE Code OF Record
As Required By
10 CFR 50.55a,
[6]

F. Jet Pumps

1. Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

4.6 SURVEILLANCE REQUIREMENTS

2. Operability testing of safety-related pumps and valves shall be performed in accordance with ~~Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f),~~ except where specific written relief has been granted by the NRC.

F. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. In the event that the jet pump(s) fail the tests in Specifications 4.6.F.1.a and 4.6.F.1.b, determine their operability by verifying that each individual jet pump $\Delta P\%$ deviation from average loop ΔP does not vary from its normal established deviation by more than 10%.

BASES: 3.6 and 4.6 (Cont'd)

throughout plant life. The inservice inspection and testing programs are performed in accordance with 10CFR50, Section 50.55a(g) except where specific relief has been granted by the NRC. These inspection and testing programs provide further assurance that gross defects are not occurring and ensure that safety-related components remain operable. [6]

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

Generic Letter 88-01 established the NRC position for in-service inspection of BWR austenitic stainless steel piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC).

2 By letter dated November 9, 1998 (NVY 98-155), NRC approved use of ASME Code Case N-560 in association with inservice inspection of Class 1, Category B-J, piping welds under ASME Section XI. VY's ASME Category B-J piping welds are also Category A piping welds as defined in GL 88-01. The Code Case reduces the inspection sample, while stipulating selection of that sample in accordance with a risk-informed analytical methodology.

The in-service inspection and testing programs ~~presented at this time~~ are based on a thorough evaluation of present technology and state-of-the-art inspection and testing techniques. [6]

F. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The following factors form the basis for the surveillance requirements:

- A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.
- The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.
- The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Specifications 4.6.F.1.a and b.

3.7 LIMITING CONDITIONS FOR OPERATION

i. Suspend movement of 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

Replace WITH TS 4.7.C.1.c

C. Secondary Containment System

1. ~~Surveillance of secondary containment shall be performed as follows:~~

   a. ~~A preoperational secondary containment capability test shall be conducted after isolating the Reactor Building and placing either Standby Gas Treatment System filter train in operation. Such tests shall demonstrate the capability to maintain a 0.15 inch of water vacuum under calm wind (2 < u < 5 mph) condition with a filter train flow rate of not more than 1500 cfm.~~ [2]

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* 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

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.

4.7 SURVEILLANCE REQUIREMENTS

~~b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.~~ [2]

1. ~~c.~~ 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 1,500 cfm, shall be demonstrated at least quarterly~~, and at each refueling outage prior to refueling~~. [3]

3.7 LIMITING CONDITIONS FOR OPERATION	4.7 SURVEILLANCE REQUIREMENTS
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.	2. Intentionally blank.
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.	3. Intentionally blank.
4. With Secondary Containment Integrity not maintained during movement of 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 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. Intentionally blank.
[8] ~~5. Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.~~	~~5. The core spray and LPCI lower compartment openings shall be checked closed daily.~~

3.10 LIMITING CONDITIONS FOR OPERATION

4. 480 V Uninterruptible Power Systems

From and after the date that one Uninterruptible Power System or its associated Motor Control Center are made or found to be inoperable for any reason, the requirements of Specification 3.5.A.4 shall be satisfied.

5. RPS Power Protection

From and after the date that one of the two redundant RPS power protection panels on an in-service RPS MG set or alternate power supply is made or found to be inoperable, the associated RPS MG set or alternate supply will be taken out of service until the panel is restored to operable status.

C. Diesel Fuel

There shall be a minimum of 36,000 usable gallons of diesel fuel in the diesel fuel oil storage tank.

4.10 SURVEILLANCE REQUIREMENTS

~~4. 480 V Uninterruptible Power Systems~~

~~When it is determined that one Uninterruptible Power System or its associated Motor Control Center is inoperable, the requirements of Specification 4.5.A.4 shall be satisfied.~~

[4]

C. Diesel Fuel

1. The quantity of diesel generator fuel shall be logged weekly and after each operation of the unit.

2. Once a month a sample of diesel fuel shall be taken and checked for quality. The quality shall be within the applicable limits specified on Table I of ASTM D975-02 and logged.

BASES: 4.10 (Cont'd)

for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The Service Discharge Test (4.10.A.2.c) is a test of the batteries ability to satisfy the design requirements of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile (battery duty cycle).

[6] Assurance that the diesels will meet their intended function is obtained by periodic surveillance testing and the results obtained from the pump and valve testing performed in accordance with the requirements of ~~ASME Section XI and~~ Specification 4.6.E. Specification 4.10.B.1.a provides an allowance to avoid unnecessary testing of the operable emergency diesel generator (EDG). If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.10.B.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, Specification 4.10.B.1.a is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of Surveillance Requirement (SR) 4.10.B.1.b suffices to provide assurance of continued operability of that EDG.

In the event the inoperable EDG is restored to operable status prior to completing either SR 4.10.B.1.a or SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1 or SR 4.10.B.3.b.2.

According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG.

Verification of operability of an off-site power source within one hour and once per eight hours thereafter as required by 4.10.B.3.b.1 may be performed as an administrative check by examining logs and other information to determine that required equipment is available and not out of service for maintenance or other reasons. It does not require performing the surveillance needed to demonstrate the operability of the equipment.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-02. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-02.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

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

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

6.2 ORGANIZATION

A. Onsite and Offsite Organizations

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

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented ~~in the~~ ~~Vermont Yankee Operational~~ Quality Assurance Program [5] Manual. The plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the Technical Requirements Manual.

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

3. The site vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

ATTACHMENT 3 TO BVY 04-118

Vermont Yankee Nuclear Power Station
Technical Specification Proposed Change No. 270
Administrative Changes

RETYPED TECHNICAL SPECIFICATION PAGES

ENTERGY NUCLEAR OPERATIONS, INC.
VERMONT YANKEE NUCLEAR POWER STATION
DOCKET NO. 50-271

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications page listed below with the revised page. The revised page contains a vertical line in the margin indicating the area of change.

Remove	Insert
Operating License – Page 3	Operating License – Page 3
98	98
110	110
120	120
121	121
143	143
155a	155a
156	156
157	157
218	218
223	223
255	255

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

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

A. Maximum Power Level

Entergy Nuclear Operations, Inc. is authorized to operate the facility at reactor core power levels not to exceed 1593 megawatts thermal in accordance with the Technical Specifications (Appendix A) appended hereto.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment 208, are hereby incorporated in the license. Entergy Nuclear Operations, Inc. shall operate the facility in accordance with the Technical Specifications.

C. Reports

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

D. This paragraph deleted by Amendment No. ____, (Date) ________.

E. Environmental Conditions

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

BASES: 3.4 & 4.4 (Cont'd)

B. Operation With Inoperable Components

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with the Requirements of Specification 4.6.E.

C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. Fixing the pump flow rate in Specification 3.4.C.3 at the minimum flow rate of 35 gpm conservatively establishes a system parameter that can be used in satisfying the ATWS requirement, as well as the original system design basis. If the product of the expression in Specification 3.4.C.3 is equal to or greater than unity, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4).

BASES:

3.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This Specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss-of-coolant analyses, the Core Spray and LPCI Systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit the accident-caused core conditions as specified in 10CFR50, Appendix K. The analyses consider appropriate combinations of the two Core Spray Subsystems and the two LPCI Subsystems associated with various break locations and equipment availability in accordance with required single failure assumptions. (Each LPCI Subsystem consists of the LPCI pumps, the recirculation pump discharge valve, and the LPCI injection valve which combine to inject torus water into a recirculation loop.)

The LPCI System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI and the Core Spray Systems provide adequate cooling for break areas up to and including the double-ended recirculation line break without assistance from the high pressure emergency Core Cooling Subsystems.

Specification 3.5.A.1 is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal with reactor pressure less than the RHR shutdown cooling permissive pressure, if capable of being manually realigned (remote) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during Hot Shutdown, if necessary.

The intent of these specifications is to prevent startup from the cold condition without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with the requirements of Specification 4.6.E.

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR System is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference: Section 14.6.3.3.2 FSAR.

Each Containment Cooling Subsystem consists of two RHR service water pumps, 1 heat exchanger, and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger, and 1 RHR pump has sufficient capacity to perform the cooling function. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with the requirements of Specification 4.6.E.

3.6 LIMITING CONDITIONS FOR OPERATION

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 150 psig and temperature greater than 350°F, both safety valves and at least three of the four relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 150 psig and 350°F within 24 hours.

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

4.6 SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

   Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter or in accordance with alternate measures approved by NRC Staff.

3.6 LIMITING CONDITIONS FOR OPERATION

F. Jet Pumps

1. Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

4.6 SURVEILLANCE REQUIREMENTS

2. Operability testing of safety-related pumps and valves shall be performed in accordance with the Code of Record as required by 10 CFR 50.55a, except where specific written relief has been granted by the NRC.

F. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. In the event that the jet pump(s) fail the tests in Specifications 4.6.F.1.a and 4.6.F.1.b, determine their operability by verifying that each individual jet pump ΔP% deviation from average loop ΔP does not vary from its normal established deviation by more than 10%.

BASES: 3.6 and 4.6 (Cont'd)

throughout plant life. The inservice inspection and testing programs are performed in accordance with 10CFR50, Section 50.55a except where specific relief has been granted by the NRC. These inspection and testing programs provide further assurance that gross defects are not occurring and ensure that safety-related components remain operable.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

Generic Letter 88-01 established the NRC position for in-service inspection of BWR austenitic stainless steel piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC).

By letter dated November 9, 1998 (NVY 98-155), NRC approved use of ASME Code Case N-560 in association with inservice inspection of Class 1, Category B-J, piping welds under ASME Section XI. VY's ASME Category B-J piping welds are also Category A piping welds as defined in GL 88-01. The Code Case reduces the inspection sample, while stipulating selection of that sample in accordance with a risk-informed analytical methodology.

The in-service inspection and testing programs are based on a thorough evaluation of present technology and state-of-the-art inspection and testing techniques.

F. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The following factors form the basis for the surveillance requirements:

- A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.
- The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.
- The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Specifications 4.6.F.1.a and b.

3.7 LIMITING CONDITIONS FOR OPERATION

i. Suspend movement of 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 1,500 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

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.

4.7 SURVEILLANCE REQUIREMENTS

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 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 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: 4.10 (Cont'd)

for the associated batteries! The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The Service Discharge Test (4.10.A.2.c) is a test of the batteries ability to satisfy the design requirements of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile (battery duty cycle).

Assurance that the diesels will meet their intended function is obtained by periodic surveillance testing and the results obtained from the pump and valve testing performed in accordance with the requirements of Specification 4.6.E. Specification 4.10.B.1.a provides an allowance to avoid unnecessary testing of the operable emergency diesel generator (EDG). If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.10.B.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, Specification 4.10.B.1.a is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of Surveillance Requirement (SR) 4.10.B.1.b suffices to provide assurance of continued operability of that EDG.

In the event the inoperable EDG is restored to operable status prior to completing either SR 4.10.B.1.a or SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1 or SR 4.10.B.3.b.2.

According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG.

Verification of operability of an off-site power source within one hour and once per eight hours thereafter as required by 4.10.B.3.b.1 may be performed as an administrative check by examining logs and other information to determine that required equipment is available and not out of service for maintenance or other reasons. It does not require performing the surveillance needed to demonstrate the operability of the equipment.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-02. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-02.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

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

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

6.2 ORGANIZATION

A. Onsite and Offsite Organizations

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

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

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

3. The site vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.