Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Vermont Yankee
185 Old Ferry Rd.
P.O. Box 500
Brattleboro, VT 05302
Tel 802-257-5271

December 7, 2004

Docket No. 50-271
BVY 04-127

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 271
<u>Removal of Inservice Testing Details</u>

References: (1) NUREG-1433, Rev. 3, Standard Technical Specifications, General Electric Plants, BWR/4, Published June 2004

In accordance with 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) hereby requests a change to Appendix A of the Vermont Yankee Nuclear Power Station Facility Operating License to:

(1) Remove Technical Specification (TS) Surveillance Requirement 4.4.A.3 for the Standby Liquid Control System pressure relief valves and revise associated bases.

(2) Remove the details of TS Surveillance Requirement 4.5.A.5 for the Recirculation pump discharge valves.

ENO has determined that this change does not involve a significant hazards consideration pursuant to 10 CFR 50.92(c).

Attachment 1 contains a description and summary safety assessment of the proposed TS change. Attachment 2 contains the determination of no significant hazards consideration in support of the proposed TS change. Attachment 3 provides a mark-up of current TS and Bases pages indicating the proposed changes. Attachment 4 provides the retyped TS and Bases pages.

ENO requests review and approval of the proposed license amendment by September 1, 2005 and a 60 day implementation period from the date of amendment approval.

There are no new commitments being made in this submittal.

If you have any questions or require additional information, please contact Mitch McCluskie at (802) 258-4187.

A047

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 7, 2004.

Sincerely,

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

Attachments:

1. Description and Assessment of Proposed Changes to the Technical Specifications Regarding Removal of Inservice Testing Details
2. No Significant Hazards Consideration
3. Mark-up of Current TS and Bases Pages
4. Retyped TS and Bases Pages

cc: Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
Mail Stop O 8 B1
Washington, DC 20555

Mr. Samuel J. Collins
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

USNRC Resident Inspector
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

ATTACHMENT 1 TO BVY 04-127

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS REGARDING REMOVAL OF INSERVICE TESTING DETAILS

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

1.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed license amendment changes Technical Specification (TS) Sections 4.4.A and 4.5.A for Vermont Yankee Nuclear Power Station (VY) regarding surveillance requirements for the Standby Liquid Control (SLC) and Low Pressure Coolant Injection (LPCI) Systems.

Specifically, the changes proposed are as follows:

1) Page 92, Surveillance Requirement (SR) 4.4.A.3: This SR is being deleted.

2) Page 97, Bases for Specifications 3.4 & 4.4: These Bases are being revised to delete discussions of pressure relief valve testing and setpoints.

3) Page 102, SR 5.4.A.5: This SR is being revised to remove details of recirculation pump discharge valve testing requirements.

2.0 BACKGROUND

ENO intends to remove the SLC pump discharge pressure setpoint requirements and the recirculation pump discharge valve closure timing requirements from the TS so that these values can be changed as necessary under the guidance of 10 CFR 50.59. The design details related to the safety functions of these systems will continue to be contained in the VY Updated Final Safety Analysis Report (UFSAR). Testing requirements and details will continue to be contained in the VY Inservice Testing (IST) Program as required by TS 4.6.E.

3.0 REASON/BASIS FOR CHANGE

Regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. This regulation requires that the TSs include items in eight specific categories. These categories are: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCO); (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

For LCOs, 10 CFR 50.36(c)(2)(ii) specifies four criteria to be used in determining whether a particular matter is required to be included in an LCO, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; (3) a structure, system, or component (SSC) that is part of the primary success path, and which functions or actuates to mitigate a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; or (4) an SSC, which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. LCOs and related requirements that fall within or satisfy any of the criteria in the regulation must be retained in the TSs, while those requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents.

The VY TSs include detailed information related to system design and testing limits. When inclusion of such information has been shown to provide little or no safety benefit, its removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of: (1) generic NRC actions; (2) new staff positions that have evolved from technological advancements and operating experience; or (3) resolution of the industry comments on Standard Technical Specifications (STSs). The NRC staff reviewed generic relaxations contained in STSs and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. ENO proposes to remove information from TS that describes engineered safety features (ESF) valve inservice testing (IST) requirements and acceptance criteria. This information is not contained in the improved STSs, NUREG-1433, Revision 3, "Standard Technical Specifications, General Electric Plants, BWR/4," dated March 31, 2004.

The safety function of the SLC pressure relief valves is to prevent damage to the SLC discharge piping due to over-pressurization, thus helping to ensure the ability of the SLC system to perform its safety function of shutting down the reactor following a failure to scram event.

The safety function of the recirculation pump discharge valves is to prevent flow from the LPCI system from being diverted from the reactor vessel, thus helping to ensure the ability of the LPCI system to perform its safety function of providing core cooling following a loss of coolant accident.

Operability testing of these components ensures their capability to perform their respective safety functions. Following implementation of the proposed change, the VY TS will still require operability testing of the subject components by reference to the VY IST Program. Details of SLC pressure relief valve and recirculation pump discharge valve testing requirements will still be contained in the VY IST Program. The SLC pressure relief valve and recirculation pump discharge valve setpoint values related to the safety functions of those systems will continue to be contained in the VY UFSAR. Changes to the VY UFSAR are evaluated per the requirements of 10 CFR 50.59. These controls are adequate to ensure the required inservice testing is performed to verify the components are operable and capable of performing their respective safety functions.

Currently, VY TS SR 5.4.A.5 specifies maximum and minimum values for the closing time of the recirculation pump discharge valves. UFSAR Section 7.4.3.5.4 includes only the maximum closing time for these valves. The analysis for the design basis loss of coolant accident assumes that the recirculation pump discharge valves close within 33 seconds to ensure that LPCI flow is directed to the reactor vessel, rather than to the postulated break in the recirculation pump suction line, such that adequate core cooling is maintained following a loss of coolant accident. Therefore, the maximum closing time of 33 seconds is an essential input to a design basis accident analysis. However, the minimum closing time of 27 seconds, which is currently specified in TS SR 5.4.A.5 has no impact on any accident or transient analysis, and therefore does not need to be controlled either in the VY TS or any VY controlled document.

NUREG-1433 includes Section 5.5.7, Inservice Testing Program, which provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. VY TS Section 4.6.E, Structural Integrity and Operability Testing, provides these same controls and is worded, in part, as follows:

> 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.

VY TS Section 4.4.A.1 states, in part:

> The Standby Liquid Control System shall be verified operable by testing pumps and valves in accordance with Specification 4.6.E.

After removal of recirculation pump discharge valve timing requirements, proposed VY TS Section 4.5.A.5 will state:

> Operability testing of recirculation pump discharge valves and bypass valves shall be in accordance with Specification 4.6.E.

Therefore, following implementation of the proposed change, the VY TS will still require operability testing of the subject components, by reference to the VY IST Program. Details of SLC pressure relief valve and recirculation pump discharge valve testing requirements are being removed from the VY TS, but are being retained in the VY IST Program. In addition, the setpoint values related to the safety functions of those systems will continue to be contained in the VY UFSAR.

In summary, the removal of these type surveillance details and specific values is acceptable because it does not alter the TS requirements that ensure the operability of the SLC and LPCI systems. In particular, the requirements of the applicable LCOs and the associated SRs for these systems, as well as the TS definition of operability, are adequate to ensure that these systems are maintained operable. As a result, these details are not necessary to ensure the SLC and LPCI systems can perform their intended safety functions and are not required to be in the TS to provide adequate protection of the public health and safety. The removal of these details is also consistent with the improved STSs, NUREG-1433. Any changes to these details will be made in accordance with 10 CFR 50.59, as specified in the VY procedures governing changes to the VY UFSAR.

4.0 ENVIRONMENTAL CONSIDERATIONS

This proposed License amendment changes SRs. VY has determined that this change involves no significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, VY has determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

ATTACHMENT 2 TO BVY 04-127

NO SIGNIFICANT HAZARDS DETERMINATION FOR
PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
REGARDING REMOVAL OF INSERVICE TESTING DETAILS

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

NO SIGNIFICANT HAZARDS DETERMINATION

The proposed license amendment changes Technical Specification (TS) Sections 4.4.A and 4.5.A for Vermont Yankee Nuclear Power Station (VY) regarding surveillance requirements for the Standby Liquid Control (SLC) and Low Pressure Coolant Injection (LPCI) Systems.

Specifically, the changes proposed are as follows:

1) Page 92, Surveillance Requirement (SR) 4.4.A.3: This SR is being deleted.

2) Page 97, Bases for Specifications 3.4 & 4.4: These Bases are being revised to delete discussions of pressure relief valve testing and setpoints.

3) Page 102, SR 5.4.A.5: This SR is being revised to remove details of recirculation pump discharge valve testing requirements.

Pursuant to 10 CFR 50.92, Entergy Nuclear Operations, Inc. (ENO) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

The proposed change does not involve a significant hazards consideration because:

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

   The proposed amendment removes details of SLC pressure relief valve and recirculation pump discharge valve testing requirements from the TS. Following implementation of the proposed change, the VY TS will still require operability testing of the subject components by reference to the VY IST Program. Details of SLC pressure relief valve and recirculation pump discharge valve testing requirements will still be contained in the VY IST Program. The SLC pressure relief valve and recirculation pump discharge valve setpoint values related to the safety functions of those systems will continue to be contained in the VY UFSAR. Changes to the VY UFSAR are evaluated per the requirements of 10 CFR 50.59. These controls are adequate to ensure the required inservice testing is performed to verify the components are operable and capable of performing their respective safety functions. The proposed amendment introduces no new equipment or changes to how equipment is operated. Neither the SLC pressure relief valves nor the recirculation pump discharge valves are initiators of any analyzed accidents. Therefore, operation of VY in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station (VY) 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 amendment removes details of SLC pressure relief valve and recirculation pump discharge valve testing requirements from the TS. The proposed amendment does not change the design or function of any component or system. No new modes of failure or initiating events are being introduced. Therefore, operation of VY in accordance with the proposed amendment 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 (VY) in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   The proposed amendment removes details of SLC pressure relief valve and recirculation pump discharge valve testing requirements from the TS. The proposed amendment does not change the design or function of any component or system. The proposed amendment does not involve any safety limits or limiting safety system settings.

   Since the proposed controls are adequate to ensure the required inservice testing is performed, there will still be high assurance that the components are operable and capable of performing their respective safety functions, and that the systems will respond as designed to mitigate the subject events. Therefore, operation of VY in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

ATTACHMENT 3 TO BVY 04-127

MARK-UP OF CURRENT TECHNICAL SPECIFICATION AND TECHNICAL SPECIFICATION BASES PAGES

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

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable when the reactor mode switch is in either the "Startup/Hot Standby" or "Run" position, except to allow testing of instrumentation associated with the reactor mode switch interlock functions provided:

1. Reactor coolant temperature is less than or equal to 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.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1320 psig shall be verified for each pump.
2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

DELETED

3. ~~Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.~~
4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor vessel is available. Both loops shall be tested over the course of two operating cycles.

BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System (SLCS) is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of natural boron in the reactor core in less than 138 minutes. An 800 ppm natural boron concentration in the reactor core is required to bring the reactor from full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% natural sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

In addition to its original design basis, the Standby Liquid Control System also satisfies the requirements of 10CFR50.62(c)(4) on anticipated transients without scram (ATWS) by using enriched boron. The ATWS rule adds hot shutdown and neutron absorber (i.e., boron-10) injection rate requirements that exceed the original Standby Liquid Control System design basis. However, changes to the Standby Liquid Control System as a result of the ATWS rule have not invalidated the original design basis.

With the reactor mode switch in the "Run" or "Startup/Hot Standby" position, shutdown capability is required. With the mode switch in "Shutdown," control rods are not able to be withdrawn since a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. With the mode switch in "Refuel," only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate shutdown margin by Specification 3.3.A ensures that the reactor will not become critical. Therefore, the Standby Liquid Control System is not required to be operable when only a single control rod can be withdrawn.

Pump operability testing (by recirculating demineralized water to the test tank) in accordance with Specification 4.6.E is adequate to detect if failures have occurred. Flow, ~~relief valve,~~ circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. The maximum SLCS pump discharge pressure during the limiting ATWS event is 1320 psig. This value is based on a peak reactor vessel lower plenum pressure of 1290 psia that occurs during the limiting ATWS event at the time of SLCS initiation, i.e., 120 seconds into the event. There is adequate margin to prevent the SLCS relief valve from lifting. ~~With a nominal SLCS relief valve setpoint of 1400 psig, there is a margin of 80 psi between the peak SLCS pump discharge pressure and the relief valve nominal setpoint.~~ Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear. In addition, at least once during each operating cycle, one of the standby liquid control loops will be initiated to verify that a flow path from a pump to the reactor vessel is available by pumping demineralized water into the reactor vessel.

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. <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

OPERABILITY TESTING OF

5. ~~R~~ecirculation pump discharge valves ~~shall be tested to verify full open to full closed in 27 ≤ t ≤ 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. Deleted.

ATTACHMENT 4 TO BVY 04-127

RETYPED TECHNICAL SPECIFICATION AND TECHNICAL SPECIFICATION BASES PAGES

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

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable when the reactor mode switch is in either the "Startup/Hot Standby" or "Run" position, except to allow testing of instrumentation associated with the reactor mode switch interlock functions provided:

1. Reactor coolant temperature is less than or equal to 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.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1320 psig shall be verified for each pump.
2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

3. Deleted

4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor vessel is available. Both loops shall be tested over the course of two operating cycles.

BASES:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System (SLCS) is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Standby Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of natural boron in the reactor core in less than 138 minutes. An 800 ppm natural boron concentration in the reactor core is required to bring the reactor from full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% natural sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

In addition to its original design basis, the Standby Liquid Control System also satisfies the requirements of 10CFR50.62(c)(4) on anticipated transients without scram (ATWS) by using enriched boron. The ATWS rule adds hot shutdown and neutron absorber (i.e., boron-10) injection rate requirements that exceed the original Standby Liquid Control System design basis. However, changes to the Standby Liquid Control System as a result of the ATWS rule have not invalidated the original design basis.

With the reactor mode switch in the "Run" or "Startup/Hot Standby" position, shutdown capability is required. With the mode switch in "Shutdown," control rods are not able to be withdrawn since a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. With the mode switch in "Refuel," only a single control rod can be withdrawn from a core cell containing fuel assemblies. Determination of adequate shutdown margin by Specification 3.3.A ensures that the reactor will not become critical. Therefore, the Standby Liquid Control System is not required to be operable when only a single control rod can be withdrawn.

Pump operability testing (by recirculating demineralized water to the test tank)in accordance with Specification 4.6.E is adequate to detect if failures have occurred. Flow, circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. The maximum SLCS pump discharge pressure during the limiting ATWS event is 1320 psig. This value is based on a peak reactor vessel lower plenum pressure of 1290 psia that occurs during the limiting ATWS event at the time of SLCS initiation, i.e., 120 seconds into the event. There is adequate margin to prevent the SLCS relief valve from lifting. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear. In addition, at least once during each operating cycle, one of the standby liquid control loops will be initiated to verify that a flow path from a pump to the reactor vessel is available by pumping demineralized water into the reactor vessel.

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. <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. Operability testing of recirculation pump discharge valves and bypass valves shall be 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. Deleted.