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October 25, 2000
BVY 00-97

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

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 238
Administrative Change

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

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

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

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued by March 2001, for implementation within 60 days of its effective date.

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

A001

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Michael A. Balduzzi
Vice President, Operations

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

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

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

SALLY A. SANDSTRUM NOTARY PUBLIC WINDHAM COUNTY, VERMONT

Attachments

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

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 238

Administrative Change

Supporting Information and Safety Assessment of Proposed Change

DESCRIPTION

This proposed change revises the Technical Specifications (TS) by correcting errors, correcting references, consolidating pages and generalizing a statement that allows NRC approved alternatives to the specified requirement. As such, these changes are administrative in nature and do not materially change any technical requirement.

The specific changes are:

(1) On TS page 24, Table 3.1.1 Notes, correct spelling errors in Note #10 and Note #12.b as follows:

- In Note #10 correct the spelling of the word "closue" to "closure"

- In Note #12.b correct the spelling of the word "faced" to "face"

(2) On TS page 83, Specification 4.3.B.3.a, remove the words "The Reactor Engineer shall."

(3) On TS page 120, Specification 4.6.E.1, split the paragraph into two paragraphs and replace the exception in the second paragraph as follows:

- Replace the words "except that sample selection for the scope of Category A welds may be in accordance with ASME Code Case N-560" with "or in accordance with alternate measures approved by NRC staff."

- Reference to ASME Code Case N-560 is added to Bases page 143 as follows:

  "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."

(4) On TS page 121, Specification 4.6.E.2, correct the reference to the Code of Federal Regulations from "50.55a(g)" to "50.55a(f)." Additionally, delete the incorrect reference provided for requesting relief in the last sentence by deleting the words "pursuant to 10 CFR 50, Section 50.55a(g) (6) (i)."

(5) On TS page 122, Specification 3.6.G.1.a, remove the hyphen from the word "Specifi-cations."

(6) On TS page 240, insert the words "Pages 241 through 252 have been intentionally deleted" and remove pages 241-252. Also, add page 240 to the Table of Contents.

(7) On TS page 256, Specification 6.2.B.7, revise the titles listed from "Operations Manager" to "Operations Superintendent," and "Assistant Operations Manager" to "Assistant Operations Superintendent."

REASON and BASES For PROPOSED CHANGES

Change #1:

This change corrects two spelling errors. There is no technical change or change in requirements involved as this is an editorial correction.

Change #2:

This change deletes the job title specified to perform a Rod Worth Minimizer verification surveillance requirement. The change does not involve a change or reduction in the frequency or requirement to perform this particular surveillance. Specifying in the surveillance, a job title for the person that performs the verification is unnecessary detail and is not contained in similar surveillance requirements nor does this type of information meet the criteria of 10CFR50.36 for inclusion in the TS. Vermont Yankee ensures that a qualified individual satisfactorily performs this task.

Change #3:

This change splits the large paragraph into two and adds a revised, general exception at the end of the second paragraph. The second paragraph, adopting GL 88-01 criteria for inservice inspection of piping susceptible to IGSCC, was added to the TS by License Amendment 139. License Amendment 172 added an option to use the selection criteria from ASME Code Case N-560. Removing the current option for ASME Code Case N-560 from this TS section will not result in any changes to requirements. Vermont Yankee (VY) has a site-specific Safety Evaluation Report (SER) authorizing use of the Code Case N-560 criteria (as an alternative to that provided in GL 88-01) at the facility and this will continue to be allowed. Thus, specifying this option is unnecessary detail in the TS. Reference to ASME Code Case N-560 approval is added to the TS Bases. Generalizing the last sentence to allow alternatives, as approved by NRC for the VY station, will allow future alternatives to be requested and approved without revising the TS surveillance wording each time. Future site specific alternatives would be requested by VY and approved by the Staff, as is the current practice. As such, this change is administrative in nature as no revision to existing technical requirements are involved and the necessity of obtaining future approval for alternatives is unchanged.

Change #4:

This change corrects the reference provided for the operability testing of safety-related pumps and valves. The correct reference should be to 10CFR50, Section 50.55a(f) in place of the specified 50.55a(g). Also proposed is the deletion of the incorrect reference to a specific section in (g) to which relief may be granted. The statement about granting of relief by the NRC is retained and only the reference to the particular 10CFR section is removed. This reference is incorrect and is unnecessary detail. Relief can be requested and granted for the reasons specified in section 50.55a(f) or other sections, as appropriate.

Change #5:

This change corrects a spelling error in that a hyphen is contained in a word inappropriately. There is no technical change in requirements involved, as this is an editorial correction.

Change #6:

This change condenses pages, where information was previously relocated by License Amendment 168. What is proposed is that a single page be retained in section 3.13 with the words that subsequent pages in that section have been deleted. No technical information currently resides on any of these pages. Page 240 will be listed in the Table of Contents also.

Change #7:

This change reflects revised titles of the unit staff. This is a change to reflect our current organization titles and no change in responsibilities or authorities of these individuals is being made.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 238

Administrative Change

Determination of No Significant Hazards Consideration

Determination of No Significant Hazards Consideration

Description of Amendment Request:

These proposed changes make editorial and administrative changes to the Technical Specifications. These changes correct spelling and grammatical errors, correct references, eliminate excessive detail related to specifying a job title, revise position titles, consolidate pages and generalize statements allowing NRC approved alternatives to specified requirements.

Basis for No Significant Hazards Determination:

Pursuant to 10CFR50.92, Vermont Yankee (VY) 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 the Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes 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 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 Technical Specification 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.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 238

Administrative Change

Marked-up Version of the Current Technical Specifications

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TABLE 3.1.1 NOTES (Cont'd)

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate ACTIONS listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 30% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. Turbine stop valve closure and turbine control valve fast ~~closue~~ scram signals may be bypassed at $\leq$30% of reactor Rated Thermal Power.

    closure

11. Not used.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

       1. Mode switch in shutdown,
       2. Manual scram,
       3. High flux IRM scram
       4. High flux SRM scram in noncoincidence,
       5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be ~~faced~~ adjacent or diagonally adjacent.

       face

3.3 LIMITING CONDITIONS FOR OPERATION

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

(a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

(b) If all rods, except those that cannot be moved with control rod drive

4.3 SURVEILLANCE REQUIREMENTS

positive coupling and the results of each test shall be recorded. The drive and blade shall be coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

(a) ~~The Reactor Engineer shall~~ verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct. <Delete>

(b) The Rod Worth Minimizer diagnostic test shall be performed.

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.

<new ¶>

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 ~~except that sample selection for the scope of Category A welds may be in accordance with ASME Code Case N-560.~~

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 Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), (f) except where specific written relief has been granted by the NRC. ~~pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).~~ <delete>

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%.

3.6 LIMITING CONDITIONS FOR OPERATION

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. <u>Single Loop Operation</u>

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

<delete>

4.6 SURVEILLANCE REQUIREMENTS

3. The surveillance requirements of 4.6.F.1 and 4.6.F.2 do not apply to the idle loop and associated jet pumps when in single loop operation.

4. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle. Baseline data for evaluating 4.6.F.2 while in single loop operation shall be updated as soon as practical after entering single loop operation.

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.

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 presented at this time 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.

~~3.13 LIMITING CONDITIONS FOR OPERATION~~

~~4.13 SURVEILLANCE REQUIREMENTS~~

Sections 3.13/4.13 deleted

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Pages 241 through 252 have been intentionally deleted

6.2 ORGANIZATION (Cont'd)

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate on-site manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Unit Staff

The unit staff organization shall include the following:

1. A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned during Plant Startup and Normal Operation.

2. At least one licensed Reactor Operator (RO) or one licensed Senior Reactor Operator (SRO) shall be present in the control room when fuel is in the reactor.

3. When the unit is in Plant Startup or Normal Operation, at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, shall be present in the control room.

4. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.B.1 and 6.2.B.8 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5. An individual qualified in radiation protection procedures shall be present on-site when there is fuel in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).

Superintendent

7. The operations ~~manager~~ or an assistant operations ~~manager~~ shall hold an SRO license.

8. While the unit is in Plant Startup or Normal Operation, the Shift Engineer shall provide advisory technical support to the Shift Supervisor (SS).

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Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 238

Administrative Change

Re-typed Technical Specification Pages

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TABLE 3.1.1 NOTES (Cont'd)

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate ACTIONS listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 30% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at $\leq$30% of reactor Rated Thermal Power.

11. Not used.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

       1. Mode switch in shutdown,
       2. Manual scram,
       3. High flux IRM scram
       4. High flux SRM scram in noncoincidence,
       5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be face adjacent or diagonally adjacent.

3.3 LIMITING CONDITIONS FOR OPERATION

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

   (a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

   (b) If all rods, except those that cannot be moved with control rod drive

4.3 SURVEILLANCE REQUIREMENTS

positive coupling and the results of each test shall be recorded. The drive and blade shall be coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

   (a) Verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

   (b) The Rod Worth Minimizer diagnostic test shall be performed.

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

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 (3.6)

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.

F. Jet Pumps (4.6)

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%.

3.6 LIMITING CONDITIONS FOR OPERATION

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Single Loop Operation

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

4.6 SURVEILLANCE REQUIREMENTS

3. The surveillance requirements of 4.6.F.1 and 4.6.F.2 do not apply to the idle loop and associated jet pumps when in single loop operation.

4. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle. Baseline data for evaluating 4.6.F.2 while in single loop operation shall be updated as soon as practical after entering single loop operation.

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.

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 presented at this time 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.

Sections 3.13/4.13 deleted

Pages 241 through 252 have been intentionally deleted.

6.2 ORGANIZATION (Cont'd)

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate on-site manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Unit Staff

The unit staff organization shall include the following:

1. A non-licensed operator shall be assigned when the reactor contains fuel and an additional non-licensed operator shall be assigned during Plant Startup and Normal Operation.

2. At least one licensed Reactor Operator (RO) or one licensed Senior Reactor Operator (SRO) shall be present in the control room when fuel is in the reactor.

3. When the unit is in Plant Startup or Normal Operation, at least one licensed Senior Reactor Operator (SRO) and one licensed Reactor Operator (RO), or two licensed Senior Reactor Operators, shall be present in the control room.

4. Shift crew composition shall meet the requirements stipulated herein and in 10 CFR 50.54(m). Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.B.1 and 6.2.B.8 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5. An individual qualified in radiation protection procedures shall be present on-site when there is fuel in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, auxiliary operators, and key maintenance personnel).

7. The operations superintendent or an assistant operations superintendent shall hold an SRO license.

8. While the unit is in Plant Startup or Normal Operation, the Shift Engineer shall provide advisory technical support to the Shift Supervisor (SS).