Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

APR 23 2001

Docket No. 50-423
B18314

RE: 10 CFR 50.90
10 CFR 50.12

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

Millstone Nuclear Power Station, Unit No. 3
Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves

Introduction

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License, NPF-49, by incorporating the attached proposed changes into the Millstone Unit No. 3 Technical Specifications. DNC is proposing to change Technical Specifications 3.1.2.1, "Reactivity Control Systems – Boration Systems – Flow Path - Shutdown;" 3.1.2.2, "Reactivity Control Systems – Flow Paths - Operating;" 3.1.2.3, "Reactivity Control Systems – Charging Pump - Shutdown;" 3.1.2.4, "Reactivity Control Systems – Charging Pumps - Operating;" 3.1.2.5, "Reactivity Control Systems – Borated Water Source - Shutdown;" 3.1.2.6, "Reactivity Control Systems – Borated Water Sources - Operating;" 3.4.1.2, "Reactor Coolant System – Hot Standby;" 3.4.1.3, "Reactor Coolant System – Hot Shutdown;" 3.4.1.4.1, "Reactor Coolant System – Cold Shutdown – Loops Filled;" 3.4.1.4.2, "Reactor Coolant System – Cold Shutdown – Loops Not Filled;" 3.4.1.6, "Reactor Coolant System – Isolated Loop Startup;" 3.4.2.1, "Reactor Coolant System – Safety Valves - Shutdown;" 3.4.2.2, "Reactor Coolant System – Operating;" 3.4.9.1, "Reactor Coolant System – Pressure/Temperature Limits;" and 3.4.9.3, "Reactor Coolant System – Overpressure Protection Systems." The Index and the associated Bases for these Technical Specifications will be modified as a result of the proposed changes.

In addition, pursuant to 10 CFR 50.12, DNC requests an exemption from the specific requirements of 10 CFR 50.60(a) and 10 CFR 50 Appendix G, based on American Society of Mechanical Engineers Code Case N-640, to support the revised reactor vessel analyses.

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Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the marked-up version of the appropriate pages of the current Technical Specifications. Attachment 4 provides the retyped pages of the Technical Specifications. Attachment 5 provides the justification for the requested exemption from 10 CFR 50.60(a) and 10 CFR 50 Appendix G.

Environmental Considerations

DNC has reviewed the proposed license amendment request against the criteria of 10 CFR 51.22 for environmental considerations. The proposed Technical Specification changes will relocate the boration subsystem and Residual Heat Removal System overpressure protection requirements (Modes 4 and 5) to a Licensee controlled document; modify the Reactor Coolant System (RCS) pressure/temperature limits; modify Cold Overpressure Protection System setpoint curves, enable temperatures and associated restrictions; modify the reactor vessel material surveillance withdrawal schedule; modify the pressurizer code safety valve requirements; modify the isolated RCS loop startup requirements; and provide numerous minor enhancements to the current requirements. The proposed changes will not adversely impact the type and amounts of effluents that may be released off site.

These changes do not result in an increase in the type and amounts of effluents that may be released off site. In addition, this amendment request will not increase individual or cumulative occupational radiation exposures. Therefore, DNC has determined the proposed changes will not have a significant effect on the quality of the human environment.

Conclusions

The proposed changes do not involve a significant impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2). Therefore, DNC requests the NRC review and approve the proposed changes to the Millstone Unit No. 3 Technical Specifications through an amendment to Operating License NPF-49, pursuant to 10 CFR 50.90.

Site Operations Review Committee and Nuclear Safety Assessment Board

The Site Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

Schedule

We request issuance of this amendment and associated exemption for Millstone Unit No. 3 prior to August 1, 2001, with the amendment to be implemented within 30 days of issuance. This will allow Millstone Unit No. 3 to use the revised RCS pressure/temperature limits before expiration of the current limits. The current pressure/temperature limits will expire approximately August 31, 2001.

State Notification

In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

Eugene S. Grecheck
Vice President - Nuclear Operations/Millstone

Sworn to and subscribed before me

this 23 day of April, 2001

Donna Lynne Williams
Notary Public

My Commission expires Nov 30, 2001

Attachments (5)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Docket No. 50-423
B18314

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Discussion of Proposed Changes and Safety Summary

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Discussion of Proposed Changes and Safety Summary

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The proposed Technical Specification changes will relocate the boration subsystem and Residual Heat Removal (RHR) System overpressurization protection requirements (Modes 4 and 5) to a Licensee controlled document; modify the Reactor Coolant System (RCS) pressure/temperature (P/T) limits; modify Cold Overpressure Protection System (COPPS) setpoint curves, enable temperatures and associated restrictions; modify the reactor vessel material surveillance withdrawal schedule; modify the pressurizer code safety valve requirements; modify the isolated RCS loop startup requirements; and provide numerous minor enhancements to the current requirements. In addition to the proposed Technical Specification changes, DNC requests an exemption to 10 CFR 50.60(a), based on American Society of Mechanical Engineers (ASME) Code Case N-640,(1) to support the revised reactor vessel analyses.

Reactor Vessel Analysis

During Refueling Outage 6, reactor vessel material surveillance capsule X was removed from the reactor vessel for subsequent evaluation and analysis. Surveillance capsule X, a standby capsule not required to be removed, was determined to meet the guidance of ASTM E 185-82.(2) It was considered to be more beneficial than waiting to remove the next scheduled capsule. The capsule specimens were destructively tested

(1) ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.

(2) ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706."

and a fluence analysis was performed which considered the actual core power history and core loading patterns. The results of this analysis was documented in WCAP-15405, Rev. 0,[3] which was submitted to the Nuclear Regulatory Commission (NRC) by a letter dated May 17, 2000.[4] This report provided relevant information for use in revising the P/T limits. Revising the P/T limits also impacts the existing COPPS evaluation and associated equipment restrictions. In addition, the withdrawal schedule will be revised to reflect removal and evaluation of surveillance capsule X and schedule changes to ensure the requirements of 10 CFR 50 Appendix H are maintained.

The current Technical Specification heatup and cooldown limitations utilize a peak vessel fluence for predicting irradiation damage to the reactor vessel beltline materials previously associated with 10 Effective Full Power Years (EFPY) of operation. The fluence used as the basis was the result of conservative fluence analyses subsequent to the first capsule evaluation (capsule U). Based on current plant performance, 10 EFPY is expected to be reached late Summer 2001. The current P/T curves utilize the guidance of ASME Section XI Appendix G, as currently required by 10 CFR 50 Appendix G.

The results of the most recent capsule removal and evaluation (capsule X) provide the most accurate results available and consider the complete power history and core loading patterns to provide revised projections of accumulated vessel fluence. The projected fluence for 32 EFPY was chosen as the basis for the proposed heatup and cooldown limitations. Consistent with the current curves and current licensing basis, Regulatory Guide 1.99, Rev. 2,[5] was used to predict the beltline material degradation. However, the development of the beltline P/T limits was established using ASME Section XI Appendix G as modified by ASME Code Case N-640. This code case, which is not currently endorsed for use by the NRC in the regulations, provides an alternate reference fracture toughness curve (K_{IC}) for establishment of the beltline P/T limits. The additional requirements of 10 CFR 50 Appendix G were considered in the establishment of the P/T limits. Clarification regarding applicablility of P/T limits to ferritic materials was also provided. The proposed changes result in less restrictive P/T limits.

The current COPPS (also referred to as low temperature overpressure protection or LTOP) setpoint curves are affected since the basis for the setpoint curves, the isothermal beltline P/T limit, has been modified. The two limiting COPPS design basis transients are the energy addition event resulting from the start of a reactor coolant pump (RCP) with the RCS secondary side 50°F higher than the RCS, and the mass addition event from a single charging pump. The setpoint curves were established to

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(3) WCAP-15405, Rev. 0, "Analysis of Capsule X from the Northeast Nuclear Energy Company Millstone Unit 3 Reactor Vessel Radiation Surveillance Program," May 2000.

(4) S. E. Scace letter to the NRC, "Millstone Nuclear Power Station, Unit No. 3 Submittal of Second Reactor Vessel Surveillance Capsule Report," dated May 17, 2000.

(5) Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

ensure the applicable limit would not be exceeded by placing operational restrictions consistent with the COPPS analyses assumptions.

The proposed COPPS setpoint curves have been established to protect the 32 EFPY isothermal reactor vessel beltline P/T curve and the power operated relief valve (PORV) discharge piping design pressure of 800 psia. When applying ASME Code Case N-640, the COPPS shall limit the maximum pressure in the vessel beltline to 100% of the isothermal beltline P/T limit curve. This condition was satisfied in the development of the COPPS setpoint curves. The limiting design basis transients associated with the COPPS setpoint curve have not changed. The limiting energy addition event remains the start of an RCP with the RCS at a temperature of no higher than 250°F, and the secondary side 50°F higher than the RCS. The limiting mass addition event remains the injection of a single charging pump (unthrottled). A staggered high and low setpoint philosophy was maintained over the entire temperature range to minimize the potential of both valves opening concurrently. Both setpoint curves are capable of protecting the P/T limit curve, which satisfies single failure assumptions. The analyses, which assume water solid operation, are conservative with respect to plant operation with a steam bubble in the pressurizer.

The primary purpose of COPPS is to provide reactor vessel overpressure protection. Currently, operational restrictions exist in the Technical Specifications to address PORV undershoot and the potential for RCP seal damage. Specifically, current restrictions prohibit the use of PORV's for COPPS protection below 160°F when an RCP is operating. Instead, the RHR System relief valves are used. Undershoot, a consequence of the PORV opening to protect the reactor vessel, occurs due to the closure stroke time of the PORV which results in RCS pressures below the opening setpoint curve. The undershoot is dependent upon the the valve size, stroke time, and the pressure transient under evaluation. Based on the proposed COPPS setpoint curves, the evaluation demonstrates that RCP seal integrity will not be challenged due to the increase in the setpoints resulting from the K_{IC} based P/T limits.

The COPPS enable temperature was previously established using the guidance of Branch Technical Position RSB 5-2.[6] The Technical Specifications currently provide a conservative value of 275°F for the COPPS enable temperature. The proposed change to a value of 226°F is based on ASME Code Section XI Appendix G (1995 Edition). The criteria is less restrictive than Branch Technical Position RSB 5-2. This approach will provide operational flexibility and enhance overall plant safety. It is endorsed by 10 CFR 50.55a.

The reactor vessel material surveillance capsule withdrawal schedule has been updated to reflect the capsule which was removed, revised lead factors, and planned capsule removal schedule. The revised schedule meets the guidance of ASTM E 185-82, which satisfies the requirements of 10 CFR 50 Appendix H.

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(6) Branch Technical Position RSB 5-2, Rev. 1, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," dated November 1988.

Use of ASME Code Case N-640

The proposed Technical Specification changes to modify the P/T limits and COPPS setpoints rely in part on the use of ASME Code Case N-640. The revised P/T limits, as specified in ASME Code Case N-640, use a higher allowable stress intensity factor, K_{IC} instead of K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor, based on the lower bound values of K_{IC} and K_{IA}. P/T curves and COPPS setpoints based on the K_{IC} curve will enhance overall plant safety by opening the P/T operating window, with the greatest safety benefit in the region of low temperature operations. In addition, enhanced safety during critical plant operational periods, heatup and cooldown evolutions, is expected.

The primary safety benefits in opening the low temperature operating window are a reduction in the challenges to pressurizer PORVs, and additional margin to maintain RCP net positive suction head (NPSH) requirements. In addition, the pressure undershoot due to the relief capacity of one PORV and the time delay for the valve to close after opening for pressure relief due to a COPPS event can result in damage to the RCP seals due to inadequate seal differential pressure. Damage to the RCP seals can require an unplanned shutdown to replace the seals. By raising the COPPS setpoints at low RCS temperatures, the likelihood of challenging the pressurizer PORVs will be reduced, and operation at higher pressures to provide additional margin for RCP seal protection will be allowed.

The P/T limits determined using ASME Code Case N-640 are less restrictive than the requirements of 10 CFR 50 Appendix G, Section IV.A.2.b, which requires the use of methods equivalent to those provided by Appendix G to ASME Section XI. Since ASME Section XI Code Case N-640 was employed in the development of the reactor vessel beltline P/T limits, an exemption to 10 CFR 50.60(a), based on ASME Code Case N-640, is required to support the proposed Technical Specification changes (Attachment 5).

Technical Specification Changes

Numerous changes to the Millstone Unit No. 3 Technical Specifications are proposed. These changes are consistent with the revised reactor vessel analyses. Additional changes have been proposed to address the relocation of the boration subsystem Technical Specifications to the Technical Requirements Manual (TRM), as well as numerous enhancements to the current requirements.

Each proposed Technical Specification change will be discussed.

Index

Changes to the Index are necessary as a result of the proposed changes to Technical Specifications 3.1.2.1, 3.1.2.2, 3.1.2.3, 3.1.2.4, 3.1.2.5, 3.1.2.6, 3.4.2.1, and 3.4.2.2, which will be discussed. The entries for these specifications on Index Page iv and vii will be replaced with the word "DELETED," as appropriate. The entry for Bases Section 3/4.1.2 on Index Page xiii will also be replaced with the word "DELETED."

Technical Specifications 3.1.2.1 through 3.1.2.6

The requirements of Technical Specifications 3.1.2.1 through 3.1.2.6 will be relocated to the TRM. The requirements contained in these specifications do not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. Refer to the Safety Summary contained in this attachment for a discussion of this criteria. The phrase "This Page Intentionally Left Blank" will be added to Pages 3/4 1-13 through 3/4 1-19.

Technical Specification 3.4.1.2

1. The Limiting Condition for Operation (LCO) will be modified by replacing the phrases "Reactor Trip System breakers are closed" and "Reactor Trip System breakers are open" with the phrases "Control Rod Drive System is capable of rod withdrawal" and "Control Rod Drive System is not capable of rod withdrawal," respectively. The proposed changes will provide operational flexibility on how to prevent rod withdrawal. It will not result in a change to the requirement for three RCS loops to be in operation when control rods can be withdrawn from the reactor.

2. Action b. will be revised by replacing the phrase "Reactor Trip System breakers in the closed position" with "Control Rod Drive System is capable of rod withdrawal." This change is consistent with the proposed LCO change.

Technical Specification 3.4.1.3

1. The LCO will be modified by replacing the phrases "Reactor Trip System breakers closed" and "Reactor Trip System breakers open" with the phrases "the Control Rod Drive System capable of rod withdrawal" and "the Control Rod Drive System not capable of rod withdrawal," respectively. The proposed changes will provide operational flexibility on how rod withdrawal is prevented. It will not result in a change to the requirement for two RCS loops to be in operation when control rods can be withdrawn from the reactor.

2. A new Action b. will be added to address less than required RCS loop operation when the Control Rod Drive System is capable of rod withdrawal. This change is consistent with the proposed LCO change.

3. Action b. will be renumbered as Action c. as a result of the addition of the new Action b. This is not a technical change.

4. The reactor coolant pump starting criteria (**) will be modified to be consistent with the revised cold overpressure protection analysis. The proposed RCP starting criteria applies to RCS loops that have not been isolated as specified in the proposed criteria and explained in the Bases for this specification. In addition, an isolated loop cannot be restored in Mode 4 in accordance with Technical Specification 3.4.1.5, "Reactor Coolant System - Isolated Loop." This will result in the following changes.

   a. Criterion "a" will be retained by specifying that the new RCP starting criteria applies to starting the first RCP.

   b. Criterion "b," which addresses energy input from the secondary system is only required when the RHR relief valves are providing cold overpressure protection. It will be retained in the revised criteria. The revised criteria will control the temperature differential between the steam generator secondary and RCS primary when RCS cold leg temperature is below the COPPS enable temperature based on the equipment providing cold overpressure protection as required by Technical Specification 3.4.9.3. The proposed criteria are consistent with the revised cold overpressure protection analysis.

   c. Criterion "c" addresses energy input from the secondary system with one RCS loop isolated and the RHR System not in service. It will be retained in the revised criteria. The revised criteria will control the temperature differential between the steam generator secondary and RCS primary when RCS cold leg temperature is below the COPPS enable temperature based on the equipment providing cold overpressure protection as required by Technical Specification 3.4.9.3. The proposed criteria are consistent with the revised cold overpressure protection analysis.

   d. Criterion "d," which addresses overpressurization of the RHR System, will not be retained. This criterion is associated with protection of the RHR System. It does not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. Refer to the Safety Summary contained in this attachment for a discussion of this criteria. This issue can be addressed by procedural controls.

5. The mass input restriction footnote (***) associated with opening the RHR System isolation valves will not be retained. This restriction is associated with protection of the RHR System. It does not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications as discussed in the Safety Summary. This issue can be addressed by procedural controls.

6. Surveillance Requirement (SR) 4.4.1.3.1 will be modified by removing the phrase "reactor coolant." This will expand the SR to also address the standby RHR pump if it is being used for compliance with the LCO.

7. SR 4.4.1.3.3 will be modified by replacing the word "loops" with "loop(s)." This is consistent with the LCO requirements which may require one or two loops to be in operation.

Technical Specification 3.4.1.4.1

1. The reactor coolant pump starting criteria (***) will be modified to be consistent with the revised cold overpressure protection analysis. The RCP starting criteria applies to RCS loops that have not been isolated as specified in the proposed criteria and explained in the Bases for this specification. An isolated loop is restored in accordance with Technical Specification 3.4.1.6. This will result in the following changes.

   a. Criterion "a" will not be retained. This criterion is associated with protection of the RCP seals due to RCS pressure undershoot following actuation of the pressurizer PORVs by the COPPS. The revised analysis has demonstrated that RCP seal integrity will not be challenged due to the increase in COPPS setpoints based on the K_{IC} P/T limits. In addition, it does not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. Refer to the Safety Summary contained in this attachment for a discussion of this criteria.

   b. Criterion "b.1" will be retained by specifying that the new RCP starting criteria applies to starting the first RCP. However, the restriction of only operating one RCP below 160°F will not be retained. This restriction, which is associated with the reactor vessel P/T analysis, will be contained in the proposed LCO of Technical Specification 3.4.9.1.

   c. Criterion "b.2" addresses energy input from the secondary system when two or more RCS loops are isolated. It will be retained in the revised criteria. In this plant configuration, only the RHR relief valves can provide cold overpressure protection as specified in the proposed LCO of Technical Specification 3.4.9.3 and explained in the associated Bases. The revised criteria will control the temperature differential between the steam generator secondary and RCS primary when the RCS is in Mode 5 (RCS cold leg temperatures below the COPPS enable temperature) based on the equipment providing cold overpressure protection (Technical Specification 3.4.9.3). The proposed criteria are consistent with the revised cold overpressure protection analysis.

   d. Criterion "b.3" addresses energy input from the secondary system with a maximum of one RCS loop isolated and the RHR System not in service. It will be retained in the revised criteria. The revised criteria will control the temperature differential between the steam generator secondary and RCS primary when the RCS is in Mode 5 (RCS cold leg temperatures below the COPPS enable temperature) based on the equipment providing cold overpressure protection (Technical Specification 3.4.9.3). The proposed criteria are consistent with the revised cold overpressure protection analysis.

e. Criterion "b.4" addresses energy input from the secondary system with a maximum of one RCS loop isolated and the RHR System in service. It will be retained in the revised criteria. The revised criteria will control the temperature differential between the steam generator secondary and RCS primary when the RCS is in Mode 5 (RCS cold leg temperatures below the COPPS enable temperature) based on the equipment providing cold overpressure protection (Technical Specification 3.4.9.3). The proposed criteria is consistent with the revised cold overpressure protection analysis.

2. SR 4.4.1.4.1.3 will be added. This SR will verify that the standby RHR pump, if required, is available. This is consistent with the LCO requirements.

<u>Technical Specification 3.4.1.4.2</u>

1. The reactor coolant pump starting criteria (***) will be modified to be consistent with the revised cold overpressure protection analysis. The RCP starting criteria applies to RCS loops that have not been isolated as specified in the proposed criteria and explained in the Bases for this specification. An isolated loop is restored in accordance with Technical Specification 3.4.1.6. This will result in the following changes.

   a. Criterion "a" will not be retained. This criteria is associated with protection of the RCP seals due to RCS pressure undershoot following actuation of the pressurizer PORVs by the COPPS. The revised analysis has demonstrated that RCP seal integrity will not be challenged due to the increase in COPPS setpoints based on the K_{IC} P/T limits. In addition, it does not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. Refer to the Safety Summary contained in this attachment for a discussion of this criteria.

   b. Criterion "b.1" will be retained by specifying that the new RCP starting criteria applies to starting the first RCP. However, the restriction of only operating one RCP below 160°F will not be retained. This restriction, which is associated with the reactor vessel P/T analysis, will be contained in the proposed LCO of Technical Specification 3.4.9.1.

c. Criterion "b.2" addresses energy input from the secondary system when two or more RCS loops are isolated. It will be retained in the revised criteria. In this plant configuration, only the RHR relief valves can provide cold overpressure protection as specified in the proposed LCO of Technical Specification 3.4.9.3 and explained in the associated Bases. The revised criteria will control the temperature differential between the steam generator secondary and RCS primary when the RCS is in Mode 5 (RCS cold leg temperatures below the COPPS enable temperature) based on the equipment providing cold overpressure protection (Technical Specification 3.4.9.3). The proposed criteria are consistent with the revised cold overpressure protection analysis.

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2. SR 4.4.1.4.2.1 will be replaced. It is not necessary to address Inservice Testing Requirements (Technical Specification 4.0.5) for these components in this specification. The RHR System pumps are already addressed by Technical Specifications 3.5.2 and 3.5.3, which require these pumps to be operable for Emergency Core Cooling. In addition, testing of the other RHR System components is already required by Technical Specification 4.0.5. The proposed SR will verify that the standby RHR pump, if required, is available. This is consistent with the LCO requirements and NUREG-1431 (Technical Specification 3.4.8).[7]

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[7] NUREG - 1431, "Standard Technical Specifications Westinghouse Plants," Revision 1, April 1995.

Technical Specification 3.4.1.6

1. The requirements of LCO 3.4.1.6 items b. and e. will be combined together. The purpose of these two requirements is to ensure opening the loop isolation valves will not result in a loss of required shutdown margin due to a reduction in RCS boron concentration to below the required value. The combined requirement to verify that the boron concentration of the isolated loop is greater than or equal to the required concentration prior to opening the loop isolation valves will ensure shutdown margin requirements will continue to be met. The 2600 ppm boron concentration limit is actually a method to ensure compliance with LCO 3.4.1.6.b. This is consistent with NUREG-1431 (Technical Specification 3.4.18 TSTF-286, Rev. 2).

2. The requirement of LCO 3.4.1.6 item c. for all RCPs to be deenergized prior to opening RCS loop isolation valves will be deleted. This requirement was added to address energy addition from the isolated loop steam generator when the loop isolation valves are opened. This requirement is not necessary since the steam generators in the RCS loops to be isolated are normally cooled down prior to closing the RCS loop isolation valves. In addition, all RCPs are administratively required to be de-energized when opening RCS loop isolation valves to minimize the differential pressure across the loop isolation valves. After the isolated loop is restored, the RCP starting criteria contained in Technical Specifications 3.4.1.4.1 and 3.4.1.4.2 would apply since an isolated RCS loop cannot be restored when the plant is in Modes 1 through 4 (Technical Specification 3.4.1.5). Guidance has been added to the Bases to address the need to verify temperature differentials if an isolated loop is restored with an RCP in operation.

3. The requirement of LCO 3.4.1.6 item d. to drain and refill the isolated loop will be deleted. This requirement is not necessary, and the associated time constraints can be a burden to plant operations. The requirements contained in LCO 3.4.1.6.b, as modified, will ensure opening the isolated loop isolation valves will not result in a loss of required shutdown margin due to a reduction in RCS boron concentration to below the required value. This requirement is actually a method to accomplish the proposed requirements of LCO 3.4.1.6.b. The need to drain and fill an isolated loop to ensure compliance with the proposed requirements of LCO 3.4.1.6.b can be adequately controlled by other administrative methods such as procedures.

4. SR 4.4.1.6.2 will be modified to be consistent with the proposed change to combine the requirements of LCO 3.4.1.6.b and e. This will ensure the SR verifies compliance with the associated LCO. In addition, SR 4.4.1.6.2 will be modified to require verification of isolated RCS loop boron concentration within 2 hours of opening the hot or cold leg stop valve instead of the current 30 minute requirement prior to opening just the cold leg stop valve.

5. SR 4.4.1.6.3 will be deleted. This is consistent with the proposed deletion of LCO 3.4.1.6.d

6. The word "Mode" will be replaced with "MODE" in the proposed LCO item b. and SR 4.4.1.6.2. This is a defined word in Technical Specifications and should be capitalized. This is a non-technical change.

Technical Specifications 3.4.2.1 and 3.4.2.2

1. Technical Specifications 3.4.2.1 and 3.4.2.2, which address the pressurizer code safety valves in Modes 1 through 4, will be combined into one Technical Specification, 3.4.2.

2. The Mode of Applicability for the new Technical Specification 3.4.2 will be reduced. The current Mode of Applicability for Technical Specification 3.4.2.2 is Modes 1 through 3. The current Mode of Applicability for Technical Specification 3.4.2.1 is Mode 4. The Mode of Applicability for the new specification will be Modes 1, 2, 3, and 4 with all RCS cold leg temperatures > 226°F. The reduction in applicability from all of Mode 4, to Mode 4 with all RCS cold leg temperatures > 226°F is consistent with the Mode of Applicability for Technical Specification 3.4.9.3, which addresses COPPS requirements. A COPPS is required to be in service when any RCS cold leg temperature is ≤ 226°F to provide overpressure protection since the pressurizer code safety valves are not adequate. Therefore, it is appropriate to limit the operability requirement for the pressurizer code safety valves to plant conditions where the COPPS is not required. This proposed change is consistent with NUREG-1431.

3. The LCO requirement for Mode 4 will be expanded to require all pressurizer code safety valves to be operable, instead of at least one pressurizer code safety valve. This more restrictive change is consistent with NUREG-1431.

4. The action requirements contained in Technical Specification 3.4.2.1 will be replaced with the requirement to be in Mode 4 with any RCS cold leg temperature ≤ 226°F. Requiring a cool down to ≤ 226°F will place the plant within the applicability of Technical Specification 3.4.9.3, which will require the COPPS to be placed in service to provide RCS overpressure protection. This is similar to the current requirement to place an operable RHR loop in service for overpressure protection. The current action requirement for Technical Specification 3.4.2.1 also requires the suspension of positive reactivity changes. Suspending positive reactivity additions in this mode of operation does not prevent the occurrence of an overpressurization event, and does not mitigate an overpressurization event. In addition, suspending positive reactivity changes would prohibit further plant cool down, assuming a negative isothermal temperature coefficient. Therefore, it would not be possible to reach the COPPS enable temperature.

The action requirements contained in Technical Specification 3.4.2.2 to address one inoperable pressurizer code safety valve will be modified. The requirement to be in Mode 4 within the following 6 hours will be changed to require Mode 4 with any RCS cold leg temperature $\leq$ 226°F within 24 hours. The plant will still be required to enter Mode 4. However, the plant will be required to be cooled down to the entry condition for Technical Specification 3.4.9.3. As a result, the completion time will be increased to provide additional time for the cooldown. An action requirement to address more than one inoperable pressurizer code safety valve will be added. If two or more pressurizer code safety valves are inoperable, the plant will be required to be in Mode 3 within 6 hours. This is more restrictive than the current action requirement which would be to apply Technical Specification 3.0.3 since this situation is not currently addressed. The action requirements of Technical Specification 3.0.3 allow an additional 1 hour to initiate action to be in Mode 3 within the next 6 hours. The proposed action requirement will not include this additional 1 hour. The proposed more restrictive action requirement is appropriate when two or more pressurizer code safety valves are inoperable.

5. SRs 4.4.2.1 and 4.4.2.2 will be combined together, and the number will be changed to 4.4.2. These are non-technical changes.

<u>Technical Specification 3.4.9.1</u>

1. The LCO of Technical Specification 3.4.9.1 will be revised to allow the deletion of the items a., b., c., and d. In addition, the LCO will specify that the limits only apply to the RCS ferritic materials. Each of these items is discussed below.

    a. Item a., which addresses RCS heatup, will be deleted. The requirement to operate in accordance with Figure 3.4-2 during RCS heatup will be contained in the revised LCO wording. In addition, the restriction on RCP operation at or below 160°F will be contained in the revised LCO. This is a non-technical change.

    b. Item b., which addresses RCS cooldown, will be deleted. The requirement to operate in accordance with Figure 3.4-3 during RCS cooldown will be contained in the revised LCO wording. In addition, the restriction of only one RCP at or below 160°F will be contained in the revised LCO. This is a non-technical change.

        The restriction of no RCP operation below 120°F will be removed. The revised analysis no longer requires this restriction.

    c. Item c., which addresses steady state RCS operation, will be deleted. The RCS pressure and temperature limits contained on Figures 3.4-2 and 3.4-3 apply at all times as specified by the applicability of this specification. However, it is not necessary to verify compliance with the limits in accordance with SR 4.4.9.1.1 during steady state operations. The Bases for this specification already contains the criteria to use to determine steady state and transient RCS operation. This guidance is used to determine when it is necessary to verify compliance with Figures 3.4-2 and 3.4-3. In addition, the restriction on RCP operation at or below 160°F will be contained in the revised LCO.

d. Item d., which addresses RCS inservice leak and hydrostatic testing operations, will be deleted. These requirements will be contained on the revised Figure 3.4-2, and the requirement to operate in accordance with this figure will be contained in the revised LCO. This is a non-technical change.

e. The LCO will be modified to specify that the limits only apply to the RCS ferritic materials. This is consistent with the requirements of 10 CFR 50 Appendix G.

2. The current action requirements for Technical Specification 3.4.9.1 will be modified. The action requirements will be separated by plant operating mode.

   a. In Modes 1 through 4, the 30 minute time period for limit restoration and the 72 hour time period for performance of the engineering evaluation will remain the same. If this evaluation is not performed in this time period, or if the evaluation does not allow continued operation, the plant will be required to enter Mode 5 ($\leq$ 200°F), instead of the current requirement to be < 200°F. This slight relaxation will have no significant impact on plant operations because plant temperature is not normally maintained at the mode change limit. Also, defining plant condition by mode, instead of by pressure and temperature, is consistent with the action requirements of most Technical Specifications. The requirement to reduce RCS pressure to below 500 psia will not change.

   b. In other than Modes 1 through 4, immediate action will be required for limit restoration. Violation of these limits is typically more severe when the RCS is cold (< 200°F), therefore an immediate response is appropriate. A time limit of prior to entering Mode 4 will be placed on the performance of the engineering evaluation. This will prevent plant startup until the evaluation has determined that the RCS is acceptable for continued operation.

   These changes are consistent with NUREG-1431.

3. Figure 3.4-2 will be replaced by a new curve, based on the revised analysis, that is applicable to 32 EFPYs. The hydrostatic and leak test limit will be added to Figure 3.4-2.

4. Figure 3.4-3 will be replaced by a new curve, based on the revised analysis, that is applicable to 32 EFPYs.

5. Table 4.4-5 will be revised. The capsule numbers and vessel locations have not been changed. The lead factors have been revised based on the evaluation of capsule X. The withdrawal column has been renamed and updated to reflect actual capsule withdrawal. A column containing fluence values has been added. The current footnote has been revised, and numerous footnotes added consistent with the proposed table changes.

Technical Specification 3.4.9.3

1. The phrase "with no more than one isolated RCS loop" will be added to LCO 3.4.9.3 items 1. and 3. This will clarify that the pressurizer PORVs cannot be used for cold overpressure protection if two or more RCS loops are isolated. This is consistent with the revised analysis.

2. The minimum required RCS vent size contained in LCO 3.4.9.3 item 44 will be reduced from 5.4 square inches to 2.0 square inches. This is consistent with the revised analysis.

3. The Mode 4 applicability will be reduced from any RCS cold leg temperature $\leq$ 275°F to $\leq$ 226°F. This is consistent with the revised analysis.

4. The Mode of Applicability footnote (**), which addresses overpressurization of the RHR System, will be deleted. This footnote is associated with protection of the RHR System. It does not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. Refer to the Safety Summary contained in this attachment for a discussion of this criteria. This issue can be addressed by procedural controls.

5. Action c. will be revised. The minimum required RCS vent size will be reduced from 5.4 square inches to 2.0 square inches to be consistent with the revised analysis. The time to establish an RCS vent will be increased from 8 hours to 12 hours. This will provide consistency between action requirements (d. and e.), and is consistent with NUREG-1431. The additional 4 hours will not result in any significant change in plant operations.

6. Action d. will be revised. The minimum required RCS vent size will be reduced from 5.4 square inches to 2.0 square inches to be consistent with the revised analysis. The time to establish an RCS vent will be increased from 8 hours to 12 hours. This will provide consistency between action requirements (c. and e.), and is consistent with NUREG-1431. The additional 4 hours will not result in any significant change in plant operations.

7. Action e. will be revised. The phrase "and with no RCS vent $\geq$ 5.4 square inches" is not necessary and will be removed. The minimum required RCS vent size will be reduced from 5.4 square inches to 2.0 square inches to be consistent with the revised analysis. The time to establish an RCS vent will be increased from 8 hours to 12 hours. This will provide consistency between action requirements (c. and d.), and is consistent with NUREG-1431. The additional 4 hours may be necessary to support compliance with this action statement since 8 hours may not be sufficient to cool down and vent the RCS if two required relief valves suddenly became inoperable.

8. SR 4.4.9.3.3 will be revised. The minimum required RCS vent size will be reduced from 5.4 square inches to 2.0 square inches to be consistent with the revised analysis.

9. Figures 3.4-4a and 3.4-4b will be replaced by new curves based on the revised analysis.

Technical Specifications Bases

The Bases for Technical Specifications 3.1.2.1, 3.1.2.2, 3.1.2.3, 3.1.2.4, 3.1.2.5, 3.1.2.6, 3.4.1.2, 3.4.1.3, 3.4.1.4.1, 3.4.1.4.2, 3.4.1.6, 3.4.2.1, 3.4.2.2, 3.4.9.1, and 3.4.9.3 will be modified as a result of the proposed Technical Specification changes. These changes are consistent with the revised analyses and the other proposed Technical Specification changes.

Additional guidance will be added to the Bases of Technical Specification 3.4.9.1 to exclude the requirement to verify compliance (SR 4.4.9.1.1) with Figures 3.4-2 and 3.4-3 when the reactor vessel is fully detensioned and to the Bases of Technical Specification 3.4.9.3 to discuss mitigation of a loss of RCS inventory or loss of shutdown margin when RCS makeup capability has been restricted to comply with COPPS mass input restrictions.

Safety Summary

Analyses Changes

The proposed revision to the Millstone Unit No. 3 P/T and COPPS curves are based on an analysis of Capsule X, a standby capsule, which was removed during Refueling Outage 6. The capsule specimens were destructively tested as required. A fluence analysis was performed which considered the actual core power history and core loading patterns.

The current Technical Specification heatup and cooldown limitations utilize a peak vessel fluence for predicting irradiation damage to the reactor vessel beltline materials previously associated with 10 EFPY of operation. In addition, the current curves utilize the guidance of ASME Section XI Appendix G, as currently required by 10 CFR 50

Appendix G.

The current COPPS enable temperature was established using the guidance of Branch Technical Position RSB 5-2. The Technical Specifications currently provide a conservative value 275°F for the COPPS enable temperature. The guidance for the proposed change to a COPPS enable temperature of 226°F is provided by ASME Code Section XI Appendix G (1995 Edition). The criteria is less restrictive than the Branch Technical Position. This approach will provide operational flexibility and enhance overall plant safety. It is endorsed by 10 CFR 50.55a.

The reactor vessel materials surveillance capsule withdrawal schedule has been updated to reflect the capsule which was removed, and to provide the revised lead factors and planned removal schedule. The revised schedule meets the guidance of ASTM E 185-82, which satisfies the requirements of 10 CFR 50 Appendix H.

The proposed changes to the P/T limits and COPPS setpoints rely in part on an exemption to 10 CFR 50.60(a), based on ASME Code Case N-640. The revised P/T limits, as specified in ASME Code Case N-640, use a higher stress intensity factor, K_{IC} instead K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor and is based on the lower bound values of K_{IC} and K_{IA}. The use of Code Case N-640 has been approved by the NRC for numerous utilities, including the Shearon Harris Nuclear Power Plant (Docket No. 400).[8]

The primary safety benefits in opening the low temperature operating window are a reduction in the challenges to pressurizer PORVs, and additional margin to maintain RCP NPSH requirements. In addition, the pressure undershoot due to the relief capacity of one PORV and the time delay for the valve to close after opening for pressure relief due to a COPPS event can result in damage to the RCP seals due to inadequate seal differential pressure. Damage to the RCP seals can require an unplanned shutdown to replace the seals. By raising the COPPS setpoint at low RCS temperatures, the likelihood of challenging the pressurizer PORVs will be reduced, and operation at higher pressures to provide additional margin for RCP seal protection will be allowed.

The proposed changes to the Technical Specification P/T curves, heatup and cooldown limitations, and the COPPS enable temperature have been developed based on the guidance contained in ASME Section XI Appendix G, as currently required by 10 CFR 50 Appendix G. In addition, the proposed chnages to the P/T curves and COPPS setpoint curves rely on an exemption to 10 CFR 50.60(a), based on ASME Code Case N-640. Since these changes are based on methodologies consistent with current industry standards, there will be no adverse affect on public safety. Therefore, the proposed changes are safe.

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[8] R. J. Laufer (NRC) letter to J. Scarola, "Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G - Shearon Harris Nuclear Power Plant, Unit 1 (TAC No. MA8643)," dated July 26, 2000.

Technical Specification Changes

The proposed Technical Specification changes have already been identified. An evaluation of the safety implications of the proposed Technical Specification changes will be presented.

Index

The proposed changes to the Index are consistent with the proposed relocation of the requirements associated with the boration subsystem to the TRM and the changes to the pressurizer code safety valve requirements. These are non-technical changes.

Relocation of Technical Specifications 3.1.2.1 through 3.1.2.6

10 CFR 50.36c(2)(ii) contains the requirements for items that must be in Technical Specifications. This regulation provides criteria that can be used to determine the requirements that must be included in the Technical Specifications. Items not meeting the criteria can be relocated from Technical Specifications to a Licensee controlled document. The Licensee can then change the relocated requirements, if necessary, in accordance with 10 CFR 50.59. This will result in significant reductions in time and expense to modify requirements that have been relocated while not adversely affecting plant safety. It is planned during the relocation of these specifications to the TRM to include changes for consistency with the other proposed Technical Specification changes (e.g., COPPS changes) contained in this submittal, and to address other previously identified enhancements. These additional changes will be evaluated in accordance with 10 CFR 50.59.

This group of Technical Specifications address the boration subsystem of the Chemical and Volume Control System. The boration subsystem is used to control the boron concentration in the RCS to maintain shutdown margin (SDM) as required by Technical Specifications 3.1.1.1.1, "Reactivity Control Systems - Boration Control Shutdown Margin - Modes 1 and 2;" 3.1.1.1.2, "Reactivity Control Systems - Boration Control Shutdown Margin - Modes 3, 4, and 5 Loops Filled;" 3.1.1.2, "Reactivity Control Systems - Shutdown Margin - Cold Shutdown - Loops Not Filled;" and 3.9.1.1, "Refueling Operations - Boron Concentration." The SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn, Technical Specification 3.1.3.5, "Reactivity Control Systems - Shutdown Rod Insertion Limit," and the control banks within the limits of Technical Specification 3.1.3.6, "Reactivity Control Systems - Control Rod Insertion Limits." When the plant is in the shutdown and refueling modes, the SDM requirements are met by adjusting RCS boron concentration.

Operation of the boration subsystem is not credited for mitigation of any Design Basis Accident (DBA) or Transient. It is assumed that the required SDM has been established prior to the start of the event. This is a valid assumption since the Technical Specification SDM requirements are required to be met prior to entering the Mode of Applicability where the event is assumed to occur. If a boron dilution event occurs in Modes 1 or 2, reactor protection is provided by the Technical Specification SDM requirements (Technical Specification 3.1.1.1.1), numerous automatic reactor trips, administrative procedures, and sufficient time for the operator to take the appropriate action (isolation of the dilution source) prior to reaching the SDM limit. If a boron dilution event occurs in Modes 3 through 6, reactor protection is provided by the Technical Specification SDM margin requirements (Technical Specifications 3.1.1.1.2, 3.1.1.2, and 3.9.1.1), administrative procedures, and sufficient time for the operator to take the appropriate action (isolation of the dilution source) prior to reaching the SDM limit. (These events are discussed in Millstone Unit No. 3 FSAR Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant.")

While the boration subsystem utilize components (e.g., charging/high head safety injection pumps and the refueling water storage tank) that also provide emergency core cooling, the Emergency Core Cooling System (ECCS) is addressed independently by Technical Specifications 3.5.2, "Emergency Core Cooling Systems – ECCS Subsystems – Tavg Greater Than or Equal to 350°F," and 3.5.3, "Emergency Core Cooling Systems – ECCS Subsystems – Tavg Less Than 350°F." Thus, the evaluation of the relocation of Technical Specifications 3.1.2.1 through 3.1.2.6 only considers the boration aspect of the affected equipment.

Technical Specifications 3.1.2.1 and 3.1.2.2 address boration subsystem flowpath requirements to ensure a flow path is available for negative reactivity control. Technical Specification 3.1.2.1 is applicable in Modes 4, 5, and 6. Technical Specification 3.1.2.2 is applicable in Modes 1, 2, and 3. A boration subsystem flowpath provides a means to supply borated water to the RCS to adjust RCS boron concentration to maintain SDM.

Technical Specifications 3.1.2.3 and 3.1.2.4 address boration subsystem charging pump requirements to ensure charging pumps are available for negative reactivity control. Technical Specification 3.1.2.3 is applicable in Modes 4, 5, and 6. Technical Specification 3.1.2.4 is applicable in Modes 1, 2, and 3. The boration subsystem charging pumps provide the motive force to supply borated water to adjust RCS boron concentration to maintain SDM.

Technical Specifications 3.1.2.5 and 3.1.2.6 address boration subsystem borated water sources to ensure a water source is available for negative reactivity control. Technical Specification 3.1.2.5 is applicable in Modes 5 and 6. Technical Specification 3.1.2.6 is applicable in Modes 1, 2, 3, and 4. The boration subsystem water sources provide the fluid source for borated water addition to adjust RCS boron concentration to maintain SDM.

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

This criterion addresses instrumentation installed to detect excessive RCS leakage. Technical Specifications 3.1.2.1 through 3.1.2.6, which ensure the boration subsystem is available for negative reactivity control, do not cover installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary. The boration subsystem does not satisfy Criterion 1.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The purpose of this criterion is to capture those process variables that have initial values assumed in the design basis accident and transient analyses, and which are monitored and controlled during power operation. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

The boration subsystem is used to establish and maintain SDM. The accident analyses assume the plant is at a specific SDM at the start of an accident. The validity of this assumption is established by the Technical Specifications that address SDM (Technical Specifications 3.1.1.1.1, 3.1.1.1.2, 3.1.1.2, and 3.9.1.1). This ensures the required SDM will be established prior to entering plant conditions (i.e., operating Mode) where the accidents are of concern. Therefore, the boration subsystem is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The boration subsystem does not satisfy Criterion 2.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The purpose of this criterion is to capture only those structures, systems, and components that are part of the primary success path of the safety analysis (an examination of the actions required to mitigate the consequences of the design basis accidents and transients). The primary success path of a safety analysis consists of the combinations and sequences of equipment needed to operate, so that the plant response to the design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. It does not include backup and diverse equipment.

The boration subsystem is used to establish and maintain SDM. The accident analyses assume the plant is at a specific SDM at the start of an accident to provide sufficient time for the plant operators to recognize the event and terminate the event prior to a complete loss of SDM. Providing sufficient time to isolate the dilution source prior to a complete loss of SDM is the primary success path for mitigation of this event. The validity of this assumption is established by the Technical Specifications that address SDM. This ensures the required SDM will be established prior to entering plant conditions where the accidents are of concern. The subsequent use of the boration subsystem to regain the required SDM is beyond the scope of a primary success path action. As a result, the boration subsystem is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The boration subsystem does not satisfy Criterion 3.

Criterion 4 A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The purpose of this criterion is to capture only those structures, systems, and components that operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Requirements proposed for relocation do not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

The boration subsystem, which is used to inject borated water to establish and maintain SDM, is not a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to the public health and safety. The boration subsystem is modeled in the Millstone Unit No. 3 Individual Plant Examination (IPE), but it has a low risk contribution since its use is limited to Anticipated Transient Without Scram (ATWS) events (beyond design basis). In addition, the emergency boration function to mitigate an ATWS event is preserved by the ECCS Technical Specification requirements (Technical Specifications 3.5.2 and 3.5.3) which require the charging pump(s) and the refueling water storage tank (RWST). A review of industry operating experience did not produce any examples where boration subsystem has had a significant adverse effect on public health and safety. The boration subsystem does not meet Criterion 4.

The requirements contained in Technical Specifications 3.1.2.1 through 3.1.2.6 for the boration subsystem do not meet the 10 CFR 50.36c(2)(ii) criteria for items that must be in Technical Specifications. Therefore, relocating these requirements from the Millstone Unit No. 3 Technical Specifications to a Licensee controlled document is safe, and will not adversely affected public health and safety.

Technical Specification 3.4.1.2

This Technical Specification addresses RCS requirements when the plant is in Mode 3. The proposed changes will provide operational flexibility on how to prevent rod withdrawal. It will not result in a change to the requirement for three RCS loops to be in operation when control rods can be withdrawn from the reactor.

Technical Specification 3.4.1.3

This Technical Specification addresses RCS requirements when the plant is in Mode 4. The proposed changes to the LCO will provide operational flexibility on how to prevent rod withdrawal. It will not result in a change to the requirement for two RCS loops to be in operation when control rods can be withdrawn from the reactor. The addition of the new action requirement is consistent with the current LCO requirements, and with similar actions already contained in Technical Specification 3.4.1.2.

Additional changes will revise the requirements that must be met prior to starting the first RCP. The revised restrictions will ensure that the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressure event beyond the capability of the COPPS to mitigate. The COPPS utilizes the pressurizer PORVs and the RHR relief valves to mitigate the limiting mass and energy addition events, thereby protecting the isothermal reactor vessel beltline. The proposed restrictions are consistent with the revised analysis, and the proposed changes to Technical Specification 3.4.9.3. Compliance with the proposed RCP starting constraints will ensure the reactor vessel will be protected from a cold overpressure event when starting the first RCP. If at least one RCP is operating, no reactor vessel protection restrictions are necessary to start additional RCPs.

The revised RCP starting criteria are based on the equipment used to provide cold overpressure protection. A maximum temperature differential of 50°F between the steam generator secondary sides and RCS cold legs will limit the potential energy addition to within the capability of the pressurizer PORVs to mitigate the transient. The RHR relief valves are also adequate to mitigate energy addition transients constrained by this temperature differential limit, provided all RCS cold leg temperatures are at or below 150 °F. The ability of the RHR relief valves to mitigate energy addition transients when RCS cold leg temperature is above 150 °F has not been analyzed. As a result, when starting an RCP with RCS cold leg temperatures above 150°F, the temperature of the steam generator secondary sides must be at or below the RCS cold leg temperatures if the RHR relief valves are providing cold overpressure protection.

RCP starting criterion "d," and the restrictions on mass input prior to placing the RHR System into service (footnote ***) are designed to prevent overpressurization of the RHR System. These restrictions will not be retained since they are associated with protection of the RHR System. The prevention of RHR System overpressurization due to starting an RCP or due to excessive mass addition does not meet any of the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. This issue can be adequately control by procedural restrictions.

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion "d" of the RCP starting requirements and the restrictions on mass input prior to placing the RHR System into service (footnote ***) contained in Technical Specification 3.4.1.3, which addresses RCS requirements when the plant is in Mode 4, do not cover installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary. These restrictions do not satisfy Criterion 1.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion "d" of the RCP starting requirements and the restrictions on mass input prior to placing the RHR System into service (footnote ***) are designed to prevent overpressurization of the RHR System. These are important restrictions for protection of the RHR System (design pressure 600 psig), but they do not challenge the integrity of a fission product barrier. Although the RHR System can be considered an extension of the RCS when the system is in service, the RHR System can be isolated by closure of the inlet and outlet motor operated valves, which are designed for full RCS pressure. Therefore, these restrictions are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The restrictions to prevent RHR System overpressurization do not satisfy Criterion 2.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion "d" of the RCP starting requirements and the restrictions on mass input prior to placing the RHR System into service (footnote ***) are designed to prevent overpressurization of the RHR System. These restrictions are important for protection of the RHR System, but they are not associated with a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design bases accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The RHR System can be isolated by closure of the inlet and outlet motor operated valves, which are designed for full RCS pressure, to mitigate a leak in the RHR System. The restrictions to prevent RHR System overpressurization do not satisfy Criterion 3.

Criterion 4 A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The RHR System is a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to the public health and safety. However, removal of the identified restrictions (Criterion "d" of the RCP starting requirements and the restrictions on mass input prior to placing the RHR System into service) will have minimal impact on the probability of a postulated RHR System overpressurization event. Therefore, removal of these restrictions will not have a significant adverse effect on public health and safety. The restrictions to prevent RHR System overpressurization do not meet Criterion 4.

The requirements contained in this specification for prevention of RHR System overpressurization due to starting an RCP or excessive mass input do not meet the 10 CFR 50.36c(2)(ii) criteria for items that must be in Technical Specifications. Therefore, removing these requirements from the Millstone Unit No. 3 Technical Specifications, and controlling them in a Licensee controlled document is safe, and will not adversely affected public health and safety.

The proposed change to SR 4.4.1.3.1 will expand the SR to address the standby RHR pump if it is being used for compliance with the LCO. This more restrictive change will provide additional assurance the requirements of Technical Specification 3.4.1.3 will be met.

The proposed change to SR 4.4.1.3.3 is consistent with the LCO requirements which may require one or two loops to be in operation. This is a non-technical change.

Technical Specification 3.4.1.4.1

This Technical Specification addresses RCS requirements when the plant is in Mode 5 with at least 2 RCS loops filled. The proposed changes will remove the restriction on RCP operation below 160°F, and revise the requirements that must be met prior to starting the first RCP. The revised RCP starting restrictions will ensure that the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressurization event beyond the capability of the COPPS to mitigate. The proposed restrictions are consistent with the revised analysis and the proposed changes to Technical Specification 3.4.9.3. Compliance with the proposed RCP starting constraints will ensure the reactor vessel will be protected from a cold overpressure event when starting the first RCP. If at least one RCP is operating, no reactor vessel protection restrictions are necessary to start additional RCPs.

The restriction on RCP operation below 160°F is associated with RCP seal protection from pressure undershoot following actuation of the pressurizer PORVs by the COPPS. Following actuation of a pressurizer PORV, RCS pressure would reach a peak and then decrease. The pressurizer PORV would receive a signal to close after RCS pressure falls below the opening setpoint. Due to the stroke time of the valve, RCS pressure would continue to fall until the valve closes. The amount by which RCS pressure falls below the setpoint has been termed undershoot. The current restriction ensures that the resultant RCS pressure due to undershoot does not fall below the minimum pressure required for RCP operation by restricting RCP operation below 160°F, the temperature point where the current PORV COPPS setpoints begin to rapidly decrease. The revised analysis, which allows for higher COPPS PORV actuation setpoints, eliminates the potential adverse impact on RCP seals due to pressure undershoot following PORV actuation. With higher actuation setpoints, the resultant pressure decrease due to undershoot is not expected to challenge the RCP seals. In addition, procedural guidance already exists to direct operator action to protect the RCPs if an excessive RCS pressure decrease occurs.

RCP operating criterion a, which is associated with protection of the RCP seals is no longer necessary due to the revised cold overpressure protection analysis. In addition, it does not meet any of the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. This issue can be addressed by procedural controls.

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion a of Technical Specification 3.4.1.4.1, which prohibits RCP operation below 160°F when the pressurizer PORVs are in service, does not cover installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary. This restriction does not satisfy Criterion 1.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion a, which restricts RCP operation below 160°F, is associated with RCP seal protection from pressure undershoot following pressurizer PORV actuation. This is important for protection of the RCP seals, but it does not represent a challenge to the integrity of a fission product barrier. RCP seal damage could result in a loss of RCS inventory, but due to the RCS conditions that would exist at this time (low temperature and low pressure), the loss of inventory would be within the RCS makeup capability established to minimize shutdown risk. In addition, the RCS loop associated with the failed RCP seal could be isolated by shutting the associated RCS loop isolation valves. Therefore, this restriction is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The restriction associated with RCP operation below 160°F does not satisfy Criterion 2.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion a, which restricts RCP operation below 160°F, is associated with RCP seal protection from pressure undershoot following pressurizer PORV actuation. This is important for protection of the RCP seals, but it is not associated with a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. RCP seal damage could result in a loss of RCS inventory, but due to the RCS conditions that would exist at this time (low temperature and low pressure), the loss of inventory would be within the RCS makeup capability established to minimize shutdown risk, and the RCS loop associated with the failed RCP seal could be isolated by shutting the associated RCS loop isolation valves. The restriction associated with RCP operation below 160°F does not satisfy Criterion 3.

Criterion 4 A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The RCPs, when the plant is shutdown, are not a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to the public health and safety. The preferred decay heat removal method when the plant is shut down is the RHR System. The RCPs are not used to meet Technical Specification requirements for decay heat removal. In addition, a review of industry operating experience did not produce any examples where failure of the RCPs when the plant is shut down has had a significant adverse effect on public health and safety. The restriction associated with RCP operation below 160°F does not meet Criterion 4.

The restriction contained in this specification for RCP seal protection from pressure undershoot following pressurizer PORV actuation does not meet the criteria of 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications. Therefore, removal of this restriction from Technical Specifications is safe, and will not adversely affect public health and safety.

RCP starting criterion b.4 and the restrictions on mass input prior to placing the RHR System into service are designed to prevent overpressurization of the RHR System. These restrictions will not be retained since they are associated with protection of the RHR System. The prevention of RHR System overpressurization due to starting an RCP, or due to excessive mass addition does not meet any of the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications (refer to criteria discussion Technical Specification 3.4.1.3). This issue can be addressed by procedural controls.

The addition of SR 4.4.1.4.1.3 to verify that the standby RHR pump, if required, is available will provide additional assurance the requirements of Technical Specification 3.4.1.4.1 will be met.

Technical Specification 3.4.1.4.2

This Technical Specification addresses RCS requirements when the plant is in Mode 5 with less than 2 RCS loops filled. The proposed changes will remove the restriction on RCP operation below 160°F and revise the requirements that must be met prior to starting the first RCP. The revised RCP starting restrictions will ensure that the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressurization event beyond the capability of the COPPS to mitigate. The proposed restrictions are consistent with the revised analysis and the proposed changes to Technical Specification 3.4.9.3. Compliance with the proposed RCP starting constraints will ensure the reactor vessel will be protected from a cold overpressure event when starting the first RCP. If at least one RCP is operating, no reactor vessel protection restrictions are necessary to start additional RCPs.

RCP operating criterion a is associated with protection of the RCP seals due to RCS pressure undershoot following actuation of the pressurizer PORVs by the COPPS. The revised analysis has eliminated the potential adverse impact on RCP seals due to pressure undershoot following PORV actuation. In addition, it does not meet any of the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications (refer to criteria discussion Technical Specification 3.4.1.4.1). This restriction can be removed from Technical Specifications.

RCP starting criterion b.4 and the restrictions on mass input prior to placing the RHR System into service are designed to prevent overpressurization of the RHR System. These restrictions will not be retained since they are associated with protection of the RHR System. The prevention of RHR overpressurization due to starting an RCP, or due to excessive mass addition does not meet any of the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications (refer to criteria discussion Technical Specification 3.4.1.3). This restriction can be addressed by procedural controls.

SR 4.4.1.4.2.1 will be replaced. It is not necessary to address Inservice Testing Requirements (Technical Specification 4.0.5) for this system in Mode 5 since this is already covered by other Technical Specifications when the plant is in Modes 1 through 4. The proposed SR will verify that the standby RHR pump, if required, is available. This is consistent with the LCO requirements and will provide additional assurance the requirements of Technical Specification 3.4.1.4.2 will be met.

Technical Specification 3.4.1.6

This Technical Specification addresses the requirements that must be met prior to opening RCS loop isolation valves when the plant is in Modes 5 or 6. The proposed

LCO item b. requirement, which consists of the combined requirements of LCO 3.4.1.6.b and e, will require verification that the boron concentration of the isolated loop is greater than or equal to the Technical Specification required SDM boron concentration prior to opening the loop isolation valves. This will ensure the required SDM is maintained.

The requirement contained in LCO item c. for all RCPs to be deenergized prior to opening RCS loop isolation valves will be deleted. This requirement is not necessary to address energy addition from the isolated loop steam generator when the loop isolation valves are opened. The loop isolation valves are normally closed after the plant reaches Mode 5 to ensure the steam generator in the RCS loop to be isolated has been cooled down prior to closing the RCS loop isolation valves. In addition, all RCPs are de-energized when opening RCS loop isolation valves to minimize the differential pressure across the loop isolation valves, and to eliminate the need to pressurize the isolated loop. If an RCP remained in operation when opening loop isolation valves, the isolated loop would have to be pressurized to match the higher RCS pressure required to operate an RCP. After the loop is unisolated, the RCP starting criteria contained in Technical Specifications 3.4.1.4.1 and 3.4.1.4.2 will apply.

The requirements of LCO item d. and the associated SR 4.4.1.6.3 to drain and refill the isolated loop will be deleted. As stated in the Bases, this requirement is designed to ensure adequate mixing and to prevent boron stratification. Adequate mixing should only be a concern if the isolated loop is drained. The methods used to refill the loop will ensure the boron concentration in the isolated loop is sufficient so that mixing is not a concern. The isolated loop is normally filled by charging with blended makeup set approximately 250 ppm above RCS boron concentration. Boron stratification will not occur at the normal operating RCS and makeup water source temperatures and boron concentrations. In addition, the boron concentration requirements contained in LCO item b. and the associated sampling requirements contained in SR 4.4.1.6.2 provide reasonable assurance that the isolated loop boron concentration is adequate. These requirements ensure that opening the isolated loop isolation valves will not result in a loss of SDM to below the required value due to a reduction in RCS boron concentration.

The requirement to drain and refill an isolated loop prior to opening the loop isolation valves was addressed in an NRC letter dated November 16, 1987.[9] Although identified in the Safety Evaluation Report contained in this letter, the need to drain and refill an isolated loop prior to restoration should be dependent on the circumstances associated with the loop isolation. If the loop is not drained after isolation, it should not have to be drained and refilled prior to restoring the loop provided the boron concentration requirements are met. Complying with the temperature and boron concentration requirements contained in the proposed Technical Specification are sufficient to ensure an excessive addition of positive reactivity does not occur. Additional procedural and administrative controls to ensure compliance with Technical

[9] R. L. Ferguson (NRC) letter to E. Mroczka, "Millstone Nuclear Power Station, Unit No. 3 - Three (N-1) Loop Operation (TAC No. 60387)," dated November 16, 1987.

Specification 3.4.1.6 can be utilized, but should not be required by Technical Specifications. This will provide operational flexibility and will not adversely affect plant safety.

The proposed changes to LCO 3.4.1.6 will result in establishing a set of requirements to control restoration of an isolated RCS loop that are consistent with generic industry guidance (NUREG-1431, Technical Specification 3.4.18 TSTF-286, Rev. 2).

The proposed change to SR 4.4.1.6.2 is consistent with the proposed change to combine the requirements of LCO 3.4.1.6.b and e. The deletion of SR 4.4.1.6.3 is consistent with the proposed deletion of LCO 3.4.1.6.d. These changes ensure SR 4.4.1.6.2 verifies compliance with the associated LCO.

The proposed change to SR 4.4.1.6.2 to require verification of isolated RCS loop boron concentration within 2 hours of opening the hot or cold leg stop valve instead of the current 30 minute requirement prior to opening just the cold leg stop valve will not result in any change to requirement contained in the LCO. The increase in time will provide better control over the loop restoration activity by allowing the requirement to be satisfied earlier in the evolution. This will allow more focus on the evolution during the immediate time period before the loop isolation valves are opened. The additional 90 minutes is not a significant enough increase in time period for the boron sample to no longer be representative of the isolated RCS loop boron concentration. In addition, the extra 90 minutes is not enough extra time to expect any other evolutions to occur that could adversely affect isolated loop boron concentration. As a result, the additional 90 minutes will not adversely impact the potential for an RCS boron reduction to below the required SDM concentration as a result of restoring the isolated RCS loop. Requiring verification of the isolated loop boron concentration before opening the hot leg or the cold leg isolation valve, instead of just before opening the cold leg isolation valve, is a more restrictive change. This will provide additional assurance an RCS boron reduction to below the required SDM concentration, as a result of restoring the isolated RCS loop, does not occur. The proposed 2 hour time period and the expansion to include the hot leg isolation valve are consistent with generic industry guidance (NUREG-1431, Surveillance Requirement 3.4.18.2).

Changing the word “Mode” to “MODE” in the proposed LCO item b and SR 4.4.1.6.2 are non-technical changes.

Technical Specifications 3.4.2.1 and 3.4.2.2

Technical Specifications 3.4.2.1 and 3.4.2.2, which address the pressurizer code safety valves in Modes 1 through 4, will be combined into one specification. The proposed Technical Specification 3.4.2 will ensure the RCS has adequate overpressure protection when operating above 226°F. If the pressurizer code safety valves are not operable, the proposed Technical Specification will require a plant shutdown that will place the plant within the capability of the COPPS to provide overpressure protection.

The proposed changes are consistent with NUREG-1431 (Technical Specification 3.4.10).

The Mode of Applicability for this new specification will be reduced slightly to be consistent with the Mode of Applicability for Technical Specification 3.4.9.3, which addresses COPPS requirements. The LCO for the pressurizer code safety valves in Mode 4 with all RCS cold leg temperatures > 226°F will be expanded to require all pressurizer code safety valves to be operable, instead of at least one pressurizer code safety valve.

The action requirement to be in Mode 4 within the following 6 hours will be changed to require Mode 4 with any RCS cold leg temperature ≤ 226°F within 24 hours. The plant will still be required to enter Mode 4. However, the plant will be required to be cooled down to the entry condition for Technical Specification 3.4.9.3. As a result, the completion time will be increased to 24 hours to provide additional time for the cooldown. This is a reasonable time period consistent with the time normally allowed to reach Mode 5 from Mode 4. Cooling the plant to 226°F approaches Mode 5 (200°F).

An action requirement to address two or more inoperable pressurizer code safety valves will be added. If two or more pressurizer code safety valves are inoperable, the plant will be required to be in Mode 3 within 6 hours. This is more restrictive than the current action requirement, which would be to apply Technical Specification 3.0.3 since this situation is not currently addressed. The action requirements of Technical Specification 3.0.3 allow an additional 1 hour to initiate action to be in Mode 3 within the next 6 hours. The proposed action requirement will not include this additional 1 hour. The proposed more restrictive action requirement is appropriate when two or more pressurizer code safety valves are inoperable.

The action requirements contained in Technical Specification 3.4.2.1 will be replaced with the requirement to be in Mode 4 with any RCS cold leg temperature ≤ 226°F. Requiring a cool down to ≤ 226°F will place the plant within the applicability of Technical Specification 3.4.9.3, which will require the COPPS to be placed in service to provide RCS overpressure protection. This is similar to the current requirement to place an operable RHR loop in service for overpressure protection. The current action requirement for Technical Specification 3.4.2.1 also requires the suspension of positive reactivity changes. Suspending positive reactivity additions in this mode of operation does not prevent the occurrence of an overpressure event, and does not mitigate an overpressure event. In addition, suspending positive reactivity changes would prohibit further plant cool down (assuming a negative isothermal temperature coefficient). Therefore, it would not be possible to reach the COPPS enable temperature, and it may be difficult to place RHR in service since Mode 4 begins at 350°F, and the RHR System is not normally placed into service just as Mode 4 is reached.

SR 4.4.2.1 will be combined with SR 4.4.2.2, and the number will be changed to 4.4.2. These are non-technical changes.

Technical Specification 3.4.9.1

This Technical Specification addresses RCS pressure and temperature restrictions designed to protect the integrity of the ferritic RCS components with the limiting component being the reactor vessel. Numerous changes have been proposed based on the evaluation of the irradiated vessel specimen recently removed.

LCO items a. and b., which address RCS heatup and cooldown, will be deleted. The requirement to operate in accordance with Figure 3.4-2 and Figure 3.4-3 during RCS heatup and cooldown will be contained in the revised LCO. In addition, the restriction on RCP operation will be contained in the revised LCO. These are non-technical changes.

LCO item b., which specifies no RCP operation below 120°F, will be removed. The revised analysis has determined this restriction is no longer necessary.

LCO item c., which addresses steady state RCS operation, will be deleted, and the restriction on RCP operation will be contained in the revised LCO. The RCS pressure and temperature limits contained on Figures 3.4-2 and 3.4-3 apply at all times as specified by the applicability of this specification. However, it is not necessary to verify compliance with the limits during steady state operations as specified in SR 4.4.9.1.1. The Bases for this specification already contains the criteria to use to determine steady state and transient RCS operation, which then determines when it is necessary to verify compliance with Figures 3.4-2 and 3.4-3. Additional guidance has been added to the Bases to also exclude the requirement to verify compliance (SR 4.4.9.1.1) with Figures 3.4-2 and 3.4-3 when the reactor vessel is fully detensioned. During refueling, with the reactor vessel head fully detensioned or removed, the RCS is not capable of being pressurized to any significant value. The limiting thermal stresses which could be encountered during this mode would be limited to flood-up using RWST water as low as 40°F. It is not possible to cause crack growth of postulated flaws in the reactor vessel at normal refueling temperatures, even if injecting 40°F water. Therefore, compliance with the requirements of Figures 3.4-2 and 3.4-3 is ensured, and verification is not required. This has been confirmed using very conservative analysis assumptions and the evaluation methods provided by ASME Section XI Appendix G.

LCO item d., which addresses RCS inservice leak and hydrostatic testing operations will be deleted. These requirements will be contained on the revised Figure 3.4-2, and the requirement to operate in accordance with this figure will be contained in the revised LCO. In addition, the restriction on RCP operation will be contained in the revised LCO. This is a non-technical change.

The LCO will be modified to specify that the limits only apply to the RCS ferritic materials. This is appropriate since the P/T limits are designed to prevent non-ductile failures, which is only applicable to the RCS ferritic materials. In addition, the Bases will be expanded to provide clarification that these curves only apply to ferritic RCS pressure boundary materials. The non-ferrritc materials comprising the balance of the reactor coolant pressure boundary are ductile over the entire RCS temperature range, and do not exhibit a transitional toughness behavior which would require lower pressures at reduced temperatures to ensure integrity. This is consistent with the requirements of 10 CFR 50 Appendix G.

The current action requirements will be modified and separated by plant operating mode. In Modes 1 through 4, the 30 minute time period for limit restoration and the 72 hour time period for performance of the engineering evaluation will remain the same. If this evaluation is not performed in this time period, or the evaluation does not allow continued operation, the plant will be required to enter Mode 5 ($\leq$ 200°F), instead of the current requirement to be < 200°F. This slight relaxation will have no significant impact on plant operations because plant temperature is not normally maintained at the mode change limit. Also, defining plant condition by mode, instead of by pressure and temperature, is consistent with the action requirements of most Technical Specifications. In other than Modes 1 through 4, immediate action will be required for limit restoration. Violation of these limits is typically more severe when the RCS is cold (< 200°F). Therefore, an immediate response is appropriate. A time limit of prior to entering Mode 4 will be placed on the performance of the engineering evaluation. This more restrictive requirement will prevent plant startup until the evaluation has determined that the RCS is acceptable for continued operation.

Figures 3.4-2 and 3.4-3 have been replaced by new curves, based on the revised analysis. The new curves are applicable to 32 EFPYs. The hydrostatic and leak test limit has been added to Figure 3.4-2.

Table 4.4-5 has been updated to reflect the most recent surveillance capsule evaluation and provides a schedule that meets ASTM E185-82 guidance. This meets the requirements of 10 CFR 50 Appendix H.

Technical Specification 3.4.9.3

This Technical Specification addresses cold overpressure protection requirements designed to protect the integrity of the reactor vessel when the RCS has been cooled down. Numerous changes have been proposed based on the evaluation of the irradiated vessel specimen recently removed.

Low temperature overpressure protection is required below the enable temperature to ensure that the P/T limits are not exceeded. Previous industry experience has demonstrated that RCS pressure can significantly exceed the allowable operating pressures provided by the P/T limits. As a result, automatic low temperature overpressure protection is required. Above the COPPS enable temperature, overpressure protection of the RCS is provided by the pressurizer code safety valves (proposed Technical Specification 3.4.2).

The Mode 4 applicability of the COPPS is established based on the enable temperature. The Mode 4 enable temperature will be reduced from any RCS cold leg temperature $\leq$ 275°F to any RCS cold leg temperature $\leq$ 226°F, consistent with the revised analysis. The current NRC guidance is provided by Branch Technical Position RSB 5-2. This guidance specifies that the COPPS be operable during startup and shutdown conditions below the enable temperature. The enable temperature is defined as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F. ASME Code Section XI Appendix G (1995 Edition) provides for an equivalent enable temperature with the exception that the metal temperature be at least RT_{NDT} + 50°F, and also provides a minimum enable temperature of 200°F. The guidance provided by ASME Code Section XI Appendix G (1995 Edition) was used to establish an enable temperature, corrected for instrument uncertainty, of 226°F. This is consistent with 10 CFR 50.55a, which accepted the use of ASME Code Section XI 1995 Edition and addenda through 1996. The proposed decrease in the enable temperature provides a larger window for plant operation.

The proposed high and low setpoint curves for the pressurizer PORV's (Figures 3.4-4a and 3.4-4b) will protect the proposed RCS P/T limits. These curves, and the other associated restrictions, ensure that should a cold overpressure event occur, the RCS P/T limits will not be exceeded.

The minimum required RCS vent size contained in LCO item 4.; action requirements c., d., and e.; and SR 4.4.9.3.3 will be reduced from 5.4 square inches to 2.0 square inches. This is consistent with the revised analysis.

The LCO will be modified to clarify that the pressurizer PORVs cannot be used for cold overpressure protection if two or more RCS loops are isolated. This is consistent with the revised analysis.

The footnote (**) associated with the applicability, which addresses overpressurization of the RHR System, will not be retained. This criteria is associated with protection of the RHR System. It does not meet the criteria contained in 10 CFR 50.36c(2)(ii) for items that must be in Technical Specifications (refer to criteria discussion Technical Specification 3.4.1.3). This issue can be addressed by procedural controls.

The time to establish an RCS vent contained in action requirements c., d., and e. will be increased from 8 hours to 12 hours. This will provide consistency between action requirements and is consistent with NUREG-1431 (Technical Specification 3.4.12 TSTF-352, Rev. 1). The additional 4 hours will not result in any significant change in plant operations. The removal of the phrase "and with no RCS vent $\geq$ 5.4 square inches" from action requirement e. will not result in any technical change.

Additional guidance will be added to the Bases to discuss mitigation of a loss of RCS inventory or loss of shutdown margin when RCS makeup capability has been restricted to comply with the COPPS mass input restrictions. The guidance specifies that it is acceptable to use additional makeup pumps, as necessary to mitigate these events, provided the respective pumps are immediately restored to a not capable of injecting status after event mitigation. This guidance, which is consistent with generic industry guidance contained in NUREG-1431 (Technical Specification 3.4.12 Bases), will not adversely impact plant safety.

Technical Specification Bases

The Bases for Technical Specifications 3.1.2.1, 3.1.2.2, 3.1.2.3, 3.1.2.4, 3.1.2.5, 3.1.2.6, 3.4.1.2, 3.4.1.3, 3.4.1.4.1, 3.4.1.4.2, 3.4.1.6, 3.4.2.1, 3.4.2.2, 3.4.9.1, and 3.4.9.3 will be modified as a result of the proposed Technical Specification changes. These changes are consistent with the revised reactor vessel analyses, and the other proposed Technical Specification changes. The additional guidance added to the Bases will ensure the requirements of the applicable Technical Specifications are applied correctly. The use of the Bases to contain information such as this is acceptable, and provides sufficient control to ensure consistency with the appropriate analyses.

Conclusion

The proposed changes to the Technical Specifications and the associated Bases are consistent with the revised reactor vessel analyses. This will ensure the analyses remain valid. In addition, the proposed changes will not result in any significant change in, or new approach to, plant operation. The proposed changes will not adversely affect public safety. Therefore, the proposed changes are safe.

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Significant Hazards Consideration

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Significant Hazards Consideration

Description of License Amendment Request

Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to revise the Millstone Unit No. 3 Technical Specifications as described in this License Amendment Request. The proposed Technical Specification changes will relocate the boration subsystem and Residual Heat Removal (RHR) System overpressurization protection requirements (Modes 4 and 5) to a Licensee controlled document; modify the Reactor Coolant System (RCS) pressure/temperature (P/T) limits; modify Cold Overpressure Protection System (COPPS) setpoint curves, enable temperatures and associated restrictions; modify the reactor vessel material surveillance withdrawal schedule; modify the pressurizer code safety valve requirements; modify the isolated RCS loop startup requirements; and provide numerous minor enhancements to the current requirements. Refer to Attachment 1 of this submittal for a discussion of the proposed changes.

Basis for No Significant Hazards Consideration

In accordance with 10 CFR 50.92, DNC has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed Technical Specification changes associated with the relocation of the boration subsystem requirements to a Licensee controlled document will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The boration function associated with this equipment is not relied upon for mitigation of any design basis event. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

   The proposed Technical Specification changes associated with the revised reactor vessel analyses will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The proposed changes extend the core burnup applicability of the Millstone Unit No. 3 RCS P/T curves, change the COPPS enable temperature and setpoint curves, and revise the reactor coolant pump (RCP) starting criteria. These revised requirements will continue to provide sufficient control over plant operation to ensure reactor vessel integrity is maintained. These restrictions do not contribute to the probability of occurrence, or consequences of, accidents previously analyzed. The revised licensing basis analyses utilize acceptable analytical methods, and continue to demonstrate that established accident analysis acceptance criteria are met. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed Technical Specification changes associated with the relocation of the Mode 4 and Mode 5 plant restrictions associated with protection of the RHR System to a Licensee controlled document will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The ability of the RHR System to remove core decay heat will not be affected. The RHR System is not relied upon for mitigation of any Mode 4 or Mode 5 design basis event. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed Technical Specification changes associated with the pressurizer code safety valves will provide additional assurance that adequate RCS overpressurization protection will be provided. The proposed changes will not affect operation of the pressurizer code safety valves, and will increase the operability requirements in Mode 4. There will be no change to safety valve reliability. The proposed changes will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed Technical Specification changes to the Modes 5 and 6 restrictions associated with restoration of an isolated RCS loop will continue to provide adequate control to prevent excessive positive reactivity additions due to water temperature or boron concentration when the isolated loop is restored to operation. Operation within the proposed restrictions will ensure a design basis accident does not occur. The design basis accidents remain the same postulated events described in the Millstone Unit No. 3 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The additional proposed changes to the Technical Specifications that will standardize terminology, relocate information to the Bases, remove extraneous information, modify the requirements to prevent rod withdrawal for operational flexibility, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional proposed changes will not increase 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.

   The proposed changes to the Technical Specifications do not impact any system or component that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The response of the plant and the operators following an accident will not be significantly different. In addition, the proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The revised analyses are based on ASME Section XI Code Case N-640, which provides an alternate reference fracture toughness curve (K_{Ic}) for establishment of the beltline P/T limits. The analyses restrictions are less restrictive than those associated with the current analyses. However, the reduction in the margin of safety is small relative to the conservatism provided by ASME Section XI margins. The analyses demonstrate that established acceptance criteria continue to be met. Specifically, the revised RCS P/T curves, COPPS enable temperature, COPPS setpoint curves, and RCP starting criteria provide acceptable margin to vessel fracture under normal operation and COPPS design basis (mass and energy additions) accident conditions. Therefore, the proposed changes will not result in a significant reduction in a margin of safety.

   The proposed Technical Specification changes associated with the relocation of the boration subsystem and RHR System overpressure protection requirements to a Licensee controlled document, pressurizer code safety valve requirements, and isolated RCS loop startup do not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed Technical Specification changes, in conjunction with administrative controls, will provide adequate control measures to ensure the accident mitigation functions will be maintained. In addition, the proposed allowed outage times and shutdown times are consistent with times already contained in the Millstone Unit No. 3 Technical Specifications and with generic industry guidance (NUREG-1431, "Standard Technical Specifications

Westinghouse Plants," Revision 1, April 1995), where applicable. Therefore, these changes will not result in a significant reduction in a margin of safety.

The additional proposed changes to the Technical Specifications that will standardize terminology, relocate information to the Bases, remove extraneous information, modify requirements to prevent rod withdrawal for operational flexibility, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional changes will not result in a significant reduction in a margin of safety.

Docket No. 50-423
B18314

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Marked Up Pages

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Marked Up Pages

Changes to the following Technical Specification pages have been proposed.

Technical Specification Section Numbers	Title(s) of Section(s)	Page and Revision Numbers
3/4.1.2 Index	Reactivity Control Systems Boration Systems	iv Amend. 99
3/4.4.2 Index	Reactor Coolant System Safety Valves	vii Amend. 193
3/4.1.2 Index	Reactivity Control Systems Boration Systems Bases	xiii Amend. 164
3/4.1.2.1	Reactivity Control Systems Boration Systems Flow Path - Shutdown	3/4 1-13 Amend. 157
3/4.1.2.2	Reactivity Control Systems Boration Systems Flow Paths- Operating	3/4 1-14 Amend. 157
3/4.1.2.3	Reactivity Control Systems Boration Systems Charging Pump - Shutdown	3/4 1-15 Amend. 157
3/4.1.2.4	Reactivity Control Systems Boration Systems Charging Pumps - Operating	3/4 1-16 Amend. 157
3/4.1.2.5	Reactivity Control Systems Boration Systems Borated Water Source - Shutdown	3/4 1-17 Amend. 113
3/4.1.2.6	Reactivity Control Systems Boration Systems Borated Water Sources - Operating	3/4 1-18 Amend. 113 3/4 1-19 Amend. 60

Technical Specification Section Numbers	Title(s) of Section(s)	Page and Revision Numbers
3/4.4.1.2	Reactor Coolant System Reactor Coolant Loops and Coolant Circulation Hot Standby	3/4 4-2 Amend. 60
3/4.4.1.3	Reactor Coolant System Reactor Coolant Loops and Coolant Circulation Hot Shutdown	3/4 4-3 Amend. 157 3/4 4-4 Amend. 145
3/4.4.1.4.1	Reactor Coolant System Reactor Coolant Loops and Coolant Circulation Cold Shutdown- Loops Filled	3/4 4-5 Amend. 157 3/4 4-5a Amend. 157
3/4.4.1.4.2	Reactor Coolant System Reactor Coolant Loops and Coolant Circulation Cold Shutdown- Loops Not Filled	3/4 4-6 Amend. 157 3/4 4-6a Amend. 157
3/4.4.1.6	Reactor Coolant System Reactor Coolant Loops and Coolant Circulation Isolated Loop Startup	3/4 4-8 Amend. 157
3/4.4.2.1	Reactor Coolant System Safety Valves - Shutdown	3/4 4-9 Amend. 106
3/4.4.2.2	Reactor Coolant System Safety Valves - Operating	3/4 4-10 Amend. 106
3/4.4.9.1	Reactor Coolant System Pressure/Temperature Limits Figure 3.4-2 Figure 3.4-3 Table 4.4-5	3/4 4-33 Amend. 157 3/4 4-34 Amend. 157 3/4 4-35 Amend. 157 3/4 4-36 Amend. 157
3/4.4.9.3	Reactor Coolant System Overpressure Protection Systems Figure 3.4-4a Figure 3.4-4b	3/4 4-38 Amend. 157 3/4 4-38a Amend. 157 3/4 4-39 Amend. 157 3/4 4-40 Amend. 157 3/4 4-41 Amend. 157

Technical Specification Section Numbers	Title(s) of Section(s)	Page and Revision Numbers
3/4.1.2	Reactivity Control Systems Boration Systems Bases	B 3/4 1-2 Amend. 157 B 3/4 1-3 Amend. 164 B 3/4 1-3a Amend. 83
3/4.4.1	Reactor Coolant System Reactor Coolant Loops and Coolant Circulation Bases	B 3/4 4-1 Amend. 157 B 3/4 4-1a Amend. 157
3/4.4.2	Reactor Coolant System Safety Valves Bases	B 3/4 4-2 Amend. 160
3/4.4.9	Reactor Coolant System Pressure/Temperature Limits Bases	B 3/4 4-7 Amend. 157 B 3/4 4-8 Amend. 157 B 3/4 4-9 Amend. 157 B 3/4 4-10 Amend. 157 B 3/4 4-11 Amend. 157 B 3/4 4-12 Amend. 157
3/4.4.9	Reactor Coolant System Overpressure Protection Systems Bases	B 3/4 4-15 Amend. 157 B 3/4 4-16 NRC Letter Dated 8/31/98 B 3/4 4-16a NRC Letter Dated 8/31/98 B 3/4 4-17 NRC Letter Dated 4/1/98 B 3/4 4-18 Amend. 157 B 3/4 4-19 Amend. 157 B 3/4 4-20 NRC Letter Dated 8/31/98 B 3/4 4-21 Amend. 157 B 3/4 4-22 NRC Letter Dated 8/31/98 B 3/4 4-23 Amend. 157 B 3/4 4-24 Amend. 157 B 3/4 4-25 Amend. 157 B 3/4 4-26 Amend. 157

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

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REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid storage system via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage system in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

a. With none of the above boron injection flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source in MODE 4, provide an OPERABLE flow path capable of being powered from an OPERABLE emergency power source within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With none of the above boron injection flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source in MODES 5 or 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°F when a flow path from the boric acid tanks is used, and

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid storage system via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limits as shown in Figure 3.1-4 at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°F when it is a required water source;

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position,

c. At least once each REFUELING INTERVAL by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

d. At least once each REFUELING INTERVAL by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 33 gpm to the RCS.

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~~REACTIVITY CONTROL SYSTEMS~~

~~CHARGING PUMP - SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.~~

~~APPLICABILITY: MODES 4, 5, and 6.~~

~~ACTION:~~

~~a. With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source in MODE 4, provide an OPERABLE charging pump capable of being powered from an OPERABLE emergency power source within 1 hour or be in COLD SHUTDOWN within the next 24 hours.~~

~~b. With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source in MODES 5 and 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that its developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.~~

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limit as shown in Figure 3.1-4 at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

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REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 6700 gallons,

2) A boron concentration between 6600 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 250,000 gallons,

2) A minimum boron concentration of 2700 ppm, and

3) A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°F.

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REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum borated water usable volume of 21,802 gallons,

2) A boron concentration between 6600 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 1,166,000 gallons,

2) A boron concentration between 2700 and 2900 ppm,

3) A minimum solution temperature of 40°F, and

4) A maximum solution temperature of 50°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least the limits as shown in Figure 3.1-4 at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

NO CHANGE
FOR INFORMATION
ONLY

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Either:

a) All reactor coolant loops shall be in operation, or

b) Three reactor coolant loops shall be in operation with THERMAL POWER restricted to less than or equal to 65% of RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

Control Rod Drive System is capable of rod withdrawal

Control Rod Drive System is not capable of rod withdrawal

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE, with at least three reactor coolant loops in operation when the ~~Reactor Trip System breakers are closed~~ or with at least one reactor coolant loop in operation when the ~~Reactor Trip System breakers are open~~:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

Control Rod Drive System is capable of rod withdrawal

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required reactor coolant loops in operation and the ~~Reactor Trip System breakers in the closed position~~, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 Either:*,**

the Control Rod Drive System capable of rod withdrawal

a. With ~~Reactor Trip System breakers closed~~, at least two RCS loops shall be OPERABLE and in operation, or

b. With ~~Reactor Trip System breakers open~~, at least two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops*** shall be OPERABLE, and at least one of these loops shall be in operation. For RCS loop(s) to be OPERABLE, at least one reactor coolant pump (RCP) shall be in operation. e

the Control Rod Drive System not capable of rod withdrawal

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

INSERT A1

~~b.~~ C. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

INSERT A2

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*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2)core outlet temperature is maintained at least 10°F below saturation temperature.

~~**A reactor coolant pump (RCP) shall not be started unless one of the following conditions is met:~~
~~a. At least one RCP is operating.~~
~~b. The secondary side water temperature of each steam generator, not isolated from the RCS, is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.~~
~~c. With a maximum of one RCS loop isolated and with the RHR relief valves isolated from the RCS, the secondary side water temperature of each steam generator, not isolated from the RCS, is less than or equal to 250°F.~~
~~d. All RCS wide range cold leg temperatures >275°F and no cold overpressure protection relief valves are in service as follows:~~
~~1) COPPS is blocked or the PORV block valves are closed, and~~
~~2) RHR relief valves are isolated from the RCS (3RHS*MV8701C or 3RHS*MV8701A is closed and 3RHS*MV8702B or 3RHS*MV8702C is closed).~~

~~***Prior to opening 3RHS*MV8701C and 3RHS*MV8701A, or 3RHS*MV8702B and 3RHS*MV8702C, all safety injection pumps and all but one centrifugal charging pump shall be incapable of injecting into the RCS. Surveillance Requirements 4.4.9.3.4 and 4.4.9.3.5 apply whenever any RHR relief valve is unisolated from the RCS.~~

INSERT A1 - Page 3/4 4-3

b. With less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

INSERT A2 - Page 3/4 4-3

** The first reactor coolant pump shall not be started when any RCS loop wide range cold leg temperature is $\leq$ 226 °F unless:

a. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature; OR

b. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required ~~reactor coolant~~ pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

4.4.1.3.3 The required ~~loops~~ loop(s) shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 17%.

APPLICABILITY: MODE 5 with at least two reactor coolant loops filled***.

---

*a. The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

INSERT B

b. All RHR loops may be removed from operation during a planned heatup to MODE 4 when at least one RCS loop is OPERABLE and in operation and when two additional steam generators are OPERABLE as required by LCO 3.4.1.4.1.b.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***a. No reactor coolant pumps (RCPs) may be in operation below 160°F unless COPPS is blocked or unless the PORV block valves are closed.

b. An RCP shall not be started unless one of the following conditions is met:
1. At least one RCP is operating and the lowest RCS wide range cold leg temperature of the unisolated RCS loops is >160°F.
2. With two or more Reactor Coolant System (RCS) loops isolated, the first RCP shall not be started unless the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
3. With a maximum of one RCS loop isolated, with the RHR relief valves isolated from the RCS, and with the PORVs providing cold overpressure protection, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
4. With a maximum of one RCS loop isolated and with any RHR relief valve unisolated from the RCS, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 200°F and less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.

INSERT B - Page 3/4 4-5

*** The first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150 °F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are $\leq$ 150 °F unless the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

ACTION:

a. With less than the required RHR loop(s) OPERABLE or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

INSERT C

INSERT C - Page 3/4 4-5a

4.4.1.4.1.3 The required pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with less than two reactor coolant loops filled***.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

INSERT D

*One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

~~*** a. No reactor coolant pumps (RCPs) may be in operation below 160°F unless COPPS is blocked or unless the PORV block valves are closed.~~

~~b. An RCP shall not be started unless one of the following conditions is met:~~

~~1. At least one RCP is operating and the lowest RCS wide range cold leg temperature of the unisolated RCS loops is >160°F.~~

~~2. With two or more Reactor Coolant System (RCS) loops isolated, the first RCP shall not be started unless the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.~~

~~3. With a maximum of one RCS loop isolated, with the RHR relief valves isolated from the RCS, and with the PORVs providing cold overpressure protection, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.~~

~~4. With a maximum of one RCS loop isolated and with any RHR relief valve unisolated from the RCS, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 200°F and less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.~~

INSERT D - Page 3/4 4-6

*** The first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150 °F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150 °F unless the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 ~~The required RHR loops shall be demonstrated OPERABLE pursuant to Specification 4.0.5.~~

4.4.1.4.2.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

INSERT E

INSERT E - Page 3/4 4-6a

The required pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

NO CHANGE
FOR INFORMATION
ONLY

LIMITING CONDITION FOR OPERATION

3.4.1.5 The RCS loop stop valves of an isolated loop shall be shut and the power removed from the valve operators.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied: either shut the loop stop valves and remove power from the valve operators within one hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.5 The RCS loop stop valves of an isolated loop shall be verified shut and power removed from the valve operators at least once per 31 days.

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REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

and

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops,

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops, or greater than 2600 ppm whichever is less,

c. All reactor coolant pumps are de-energized.

d. The isolated portion of the loop has been drained and is refilled, and

e. The reactor is subcritical by at least the value required by Specifications 3.1.1.1.2 or 3.1.1.2 for ~~Mode~~ 5 or Specification 3.9.1.1 for ~~Mode~~ 6.

MODE

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

MODE hot or 2 hours

4.4.1.6.2 ~~The reactor shall be determined to be subcritical by at least the value~~ required by Specifications 3.1.1.1.2 or 3.1.1.2 for ~~Mode~~ 5 or Specification 3.9.1.1 for ~~Mode~~ 6 within ~~30 minutes~~ prior to opening the cold leg stop valve.

MODE

4.4.1.6.3 Within 4 hours prior to opening the loop stop valves, the isolated loop shall be determined to:

a. Be drained and refilled, and

b. Have a boron concentration greater than or equal to the boron concentration of the operating loops, or greater than 2600 ppm whichever is less.

The isolated loop boron concentration shall be determined to be greater than or equal to the boron concentration

March 17, 1995

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

~~SHUTDOWN~~

LIMITING CONDITION FOR OPERATION

All

5

3.4.2~~.1~~ ~~A minimum of one~~ pressurizer Code safety valve shall be OPERABLE with a lift setting* of 2500 psia ± 3%.**

APPLICABILITY: ~~MODE 4.~~ MODES 1, 2, and 3, MODE 4 with all RCS cold leg temperatures >226 °F.

ACTION:

~~With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.~~

INSERT F

SURVEILLANCE REQUIREMENTS

4.4.2~~.1~~ No additional Surveillance Requirements other than those required by Specification 4.0.5.

---

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**The lift setting shall be within ± 1% following pressurizer Code safety valve testing.

INSERT F - Page 3/4 4-9

With one pressurizer Code safety valve inoperable, restore the inoperable valve to OPERABLE status within 15 minutes. If the inoperable valve is not restored to OPERABLE status within 15 minutes, or if two or more pressurizer Code safety valves are inoperable, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with any RCS cold leg temperature $\leq$ 226 °F within the following 24 hours.

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March 17, 1995

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2500 psia ± 3%.**

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**The lift setting shall be within ± 1% following pressurizer Code safety valve testing.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

INSERT 61

LIMITING CONDITION FOR OPERATION

3.4.9.1 The reactor coolant system (except the pressurizer) temperature and pressure shall be limited as follows:

a. During an RCS heatup, the heatup limits of Figure 3.4-2 apply with the additional restriction that only one reactor coolant pump can be operating when the lowest unisolated RCS loop wide range cold leg temperature is $\leq$160°F.

b. During an RCS cooldown, the limits of Figure 3.4-3 apply with the additional restriction that only one reactor coolant pump can be operating when the lowest unisolated RCS loop wide range cold leg temperature is $\leq$160°F and no reactor coolant pump may be operated when the lowest unisolated RCS loop wide range cold leg temperature is $\leq$120°F.

c. During steady state conditions, when the maximum temperature increase or decrease in any one hour period is <10°F and when the plant is not changing temperatures in accordance with a heatup or cooldown procedure, only one reactor coolant pump can be operating when the lowest unisolated RCS loop wide range cold leg temperature is $\leq$160°F. The limits of Figures 3.4-2 and 3.4-3 do not apply during steady state conditions.

d. During RCS inservice leak and hydrostatic testing operations, the Hydrostatic and Leak Test limit of Figure 3.4-2 apply with the additional restrictions that within a one-hour period prior to exceeding the heatup curve, and during each one-hour period above the heatup curve, a maximum temperature increase or decrease of 5°F in any one-hour period is allowed.

APPLICABILITY: At all times.

INSERT 62

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup and cooldown operations, and during the one-hour period prior to and during inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3 as required.

INSERT G1 - Page 3/4 4-33

Reactor Coolant System (except the pressurizer) temperature, pressure, and heatup and cooldown rates of ferritic materials shall be limited in accordance with the limits shown on Figures 3.4-2 and 3.4-3. In addition, a maximum of one reactor coolant pump can be in operation when the lowest unisolated Reactor Coolant System loop wide range cold leg temperature is $\leq$ 160 °F.

INSERT G2 - Page 3/4 4-33

a. With any of the above limits exceeded in MODES 1, 2, 3, or 4, perform the following:

    1. Restore the temperature and/or pressure to within limit within 30 minutes.

    AND

    2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System remains acceptable for continued operation within 72 hours. Otherwise, be in at least MODE 3 within the next 6 hours and in MODE 5 with RCS pressure less than 500 psia within the following 30 hours.

b. With any of the above limits exceeded in other than MODES 1, 2, 3, or 4, perform the following:

    1. Immediately initiate action to restore the temperature and/or pressure to within limit.

    AND

    2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System is acceptable for continued operation prior to entering MODE 4.

INSERT New Figure 3.4-2

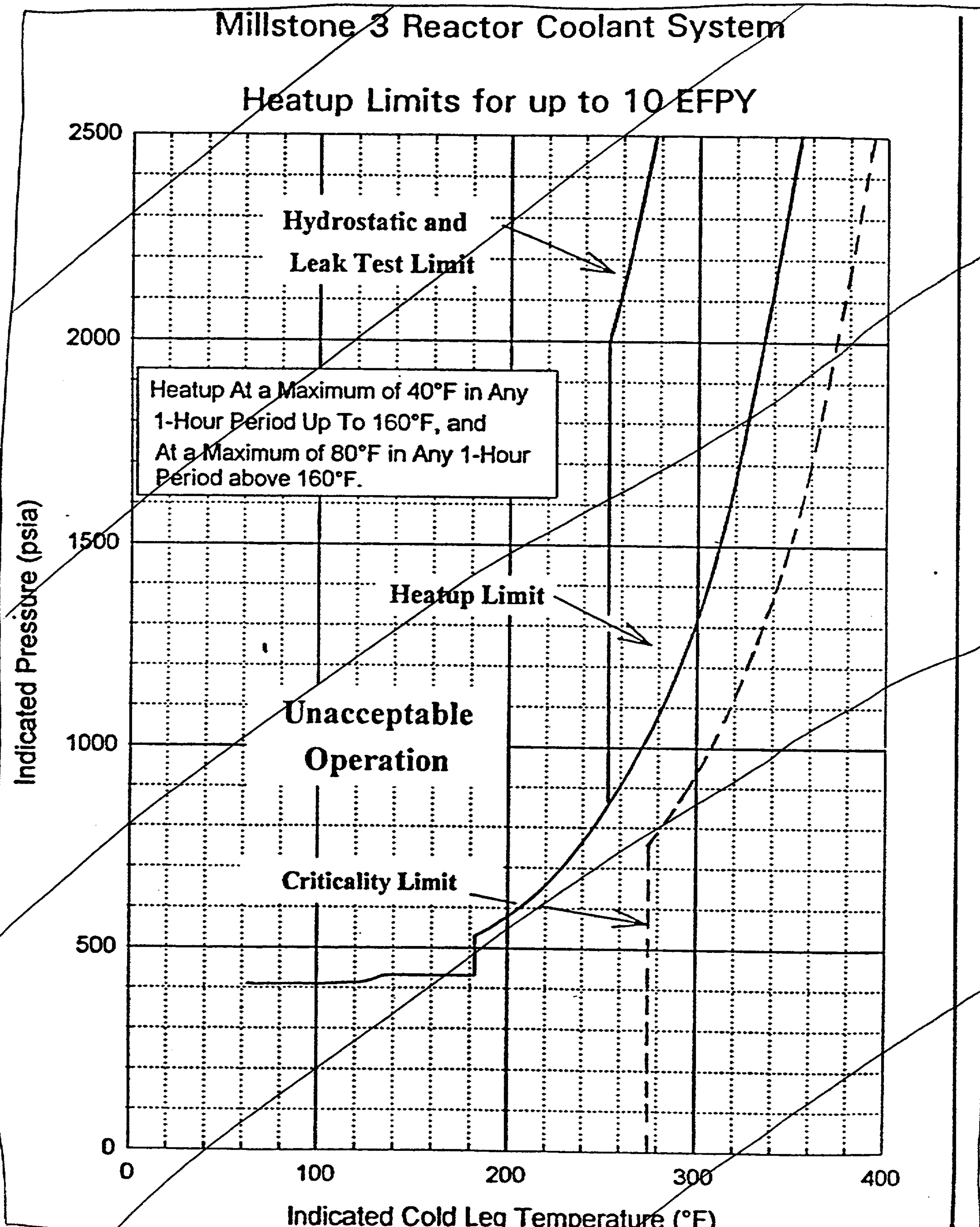
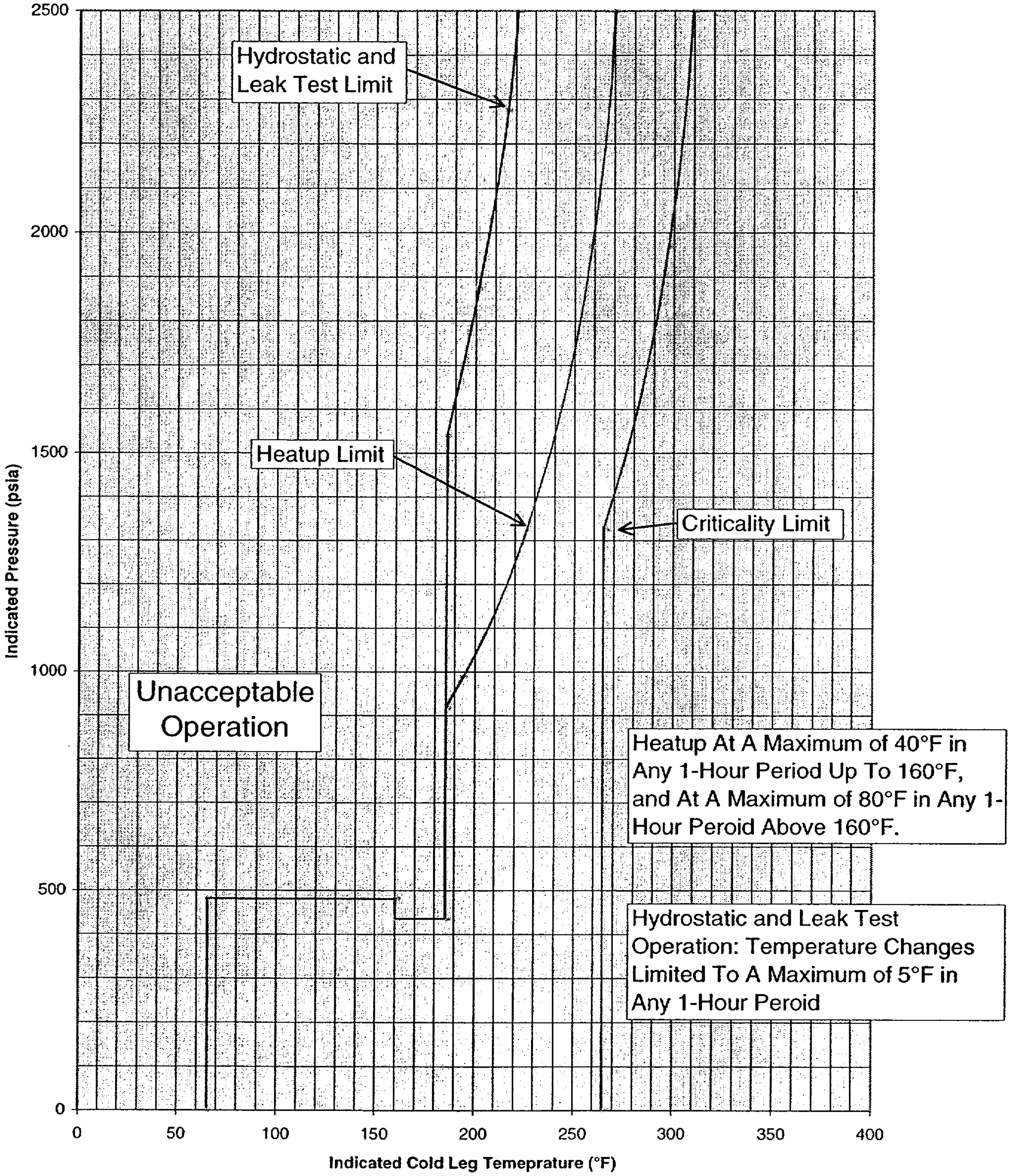


FIGURE 3.4-2

Millstone 3 Reactor Coolant System
Heatup Limitations for Fluence up to 1.97E+19 n/cm (32 EFPY)



INSERT New Figure 3.4-3 2/12/98

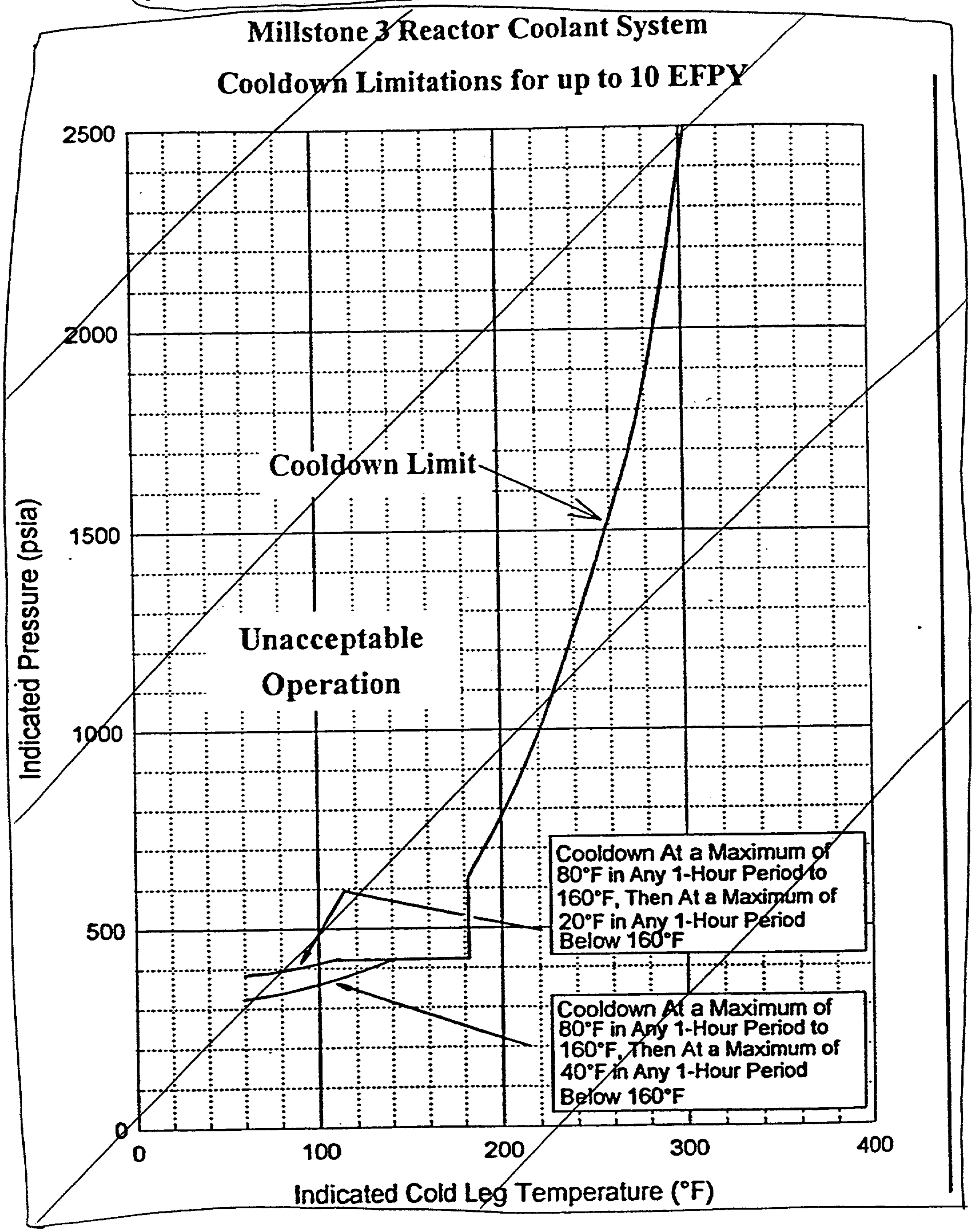


Figure 3.4-3

Millstone 3 Reactor Coolant System
Cooldown Limitations for Fluence up to 1.97E+19 n/cm (32 EFPY)

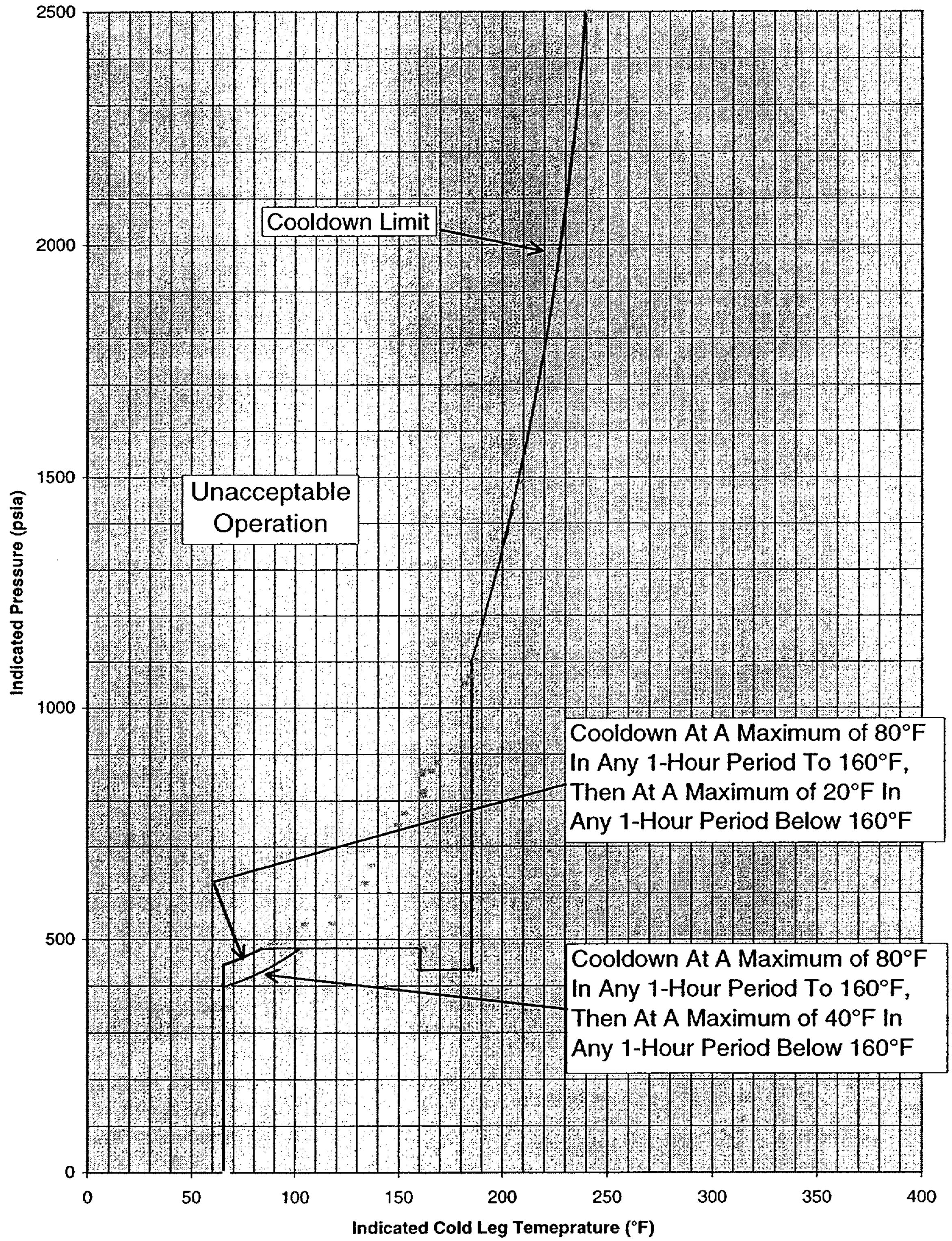


TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	APPROXIMATE WITHDRAWAL TIME (EFPY)
U	58.5°	3.98(a)	First Refueling (1.3 EFPY actual)
Y	241°	3.74	9
V	61°	3.74	16
W	121.5°	4.01	STANDBY
X	238.5°	4.01	STANDBY
Z	301.5°	4.01	STANDBY

a) Plant specific evaluation.

INSERT New Table 4.4-5

TABLE 4.4-5

Millstone Unit 3 Reactor Vessel Surveillance Capsule Withdrawal Schedule

CAPSULE	LOCATION	LEAD FACTOR[a]	REMOVAL TIME (EFPY)[b]	FLUENCE (n/cm^2, E>1.0MeV)[a]
U	58.5°	4.31	1.3	4.49×10^{18}(c)
X	238.5°	4.37	8.0	2.21×10^{19}(c)
W	121.5°	4.32	Approx. 14.0	3.76×10^{19}(c,d)
Y[e]	241°	4.11	Standby	
V[e]	61°	4.11	Standby	
Z[e]	301.5°	4.32	Standby	

(a) Updated in Capsule X dosimetry analysis.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This projected fluence is not less than once or greater than twice the peak end of license EOL fluence, and is approximately equal to the peak vessel fluence at 54 EFPY.

(e) These capsules will be at the approximate 54 EFPY peak surface (i.e. clad/base metal interface) fluence when capsule W is withdrawn.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Cold Overpressure Protection shall be OPERABLE with a maximum of one centrifugal charging pump* and no Safety Injection pumps capable of injecting into the Reactor Coolant System (RCS) and one of the following pressure relief capabilities:

1. One power operated relief valve (PORV) with a nominal lift setting established in Figure 3.4-4a and one PORV with a nominal lift setting established in Figure 3.4-4b, or

with no more than one isolated RCS loop

2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 426.8 psig and ≤ 453.2 psig, or

3. One PORV with a nominal lift setting established in Figure 3.4-4a or Figure 3.4-4b and one RHR suction relief valve with a setpoint ≥ 426.8 psig and ≤ 453.2 psig, or

4. RCS depressurized with an RCS vent of ≥ ~~5.4~~ 2.0 square inches.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~275~~ 226°F~~**~~, MODE 5, and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With two or more centrifugal charging pumps capable of injecting into the RCS, immediately initiate action to establish that a maximum of one centrifugal charging pump is capable of injecting into the RCS.

b. With any Safety Injection pump capable of injecting into the RCS, immediately initiate action to establish that no Safety Injection pumps are capable of injecting into the RCS.

c. With one required relief valve inoperable in MODE 4, restore the required relief valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least a ~~5.4~~ 2.0 square inch vent within the next ~~8~~ 12 hours.

---

*Two centrifugal charging pumps may be capable of injecting into the RCS for less than one hour, during pump swap operations. However, at no time will two charging pumps be simultaneously out of pull-to-lock during pump swap operations.

~~**When an RHR suction relief valve or a PORV, which is armed for COPPS, is unisolated from the RCS, and when the RCS cold leg temperature is greater than 275°F, a maximum of one centrifugal charging pump and no safety injection pumps shall be capable of injecting into the RCS.~~

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

d. With one required relief valve inoperable in MODE 5 or 6, restore the required relief valve to OPERABLE status within 24 hours, or depressurize the RCS and establish an RCS vent of ≥ ~~5.4~~ 2.0 square inches within the next ~~8~~ 12 hours.

e. With two required relief valves inoperable ~~and with no RCS vent ≥ 5.4 square inches~~, depressurize the RCS and establish an RCS vent of ≥ ~~5.4~~ 2.0 square inches within ~~8~~ 12 hours.

f. In the event the PORVs, the RHR suction relief valves, or the RCS vent are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHR suction relief valves, or RCS vent on the transient, and any corrective action necessary to prevent recurrence.

g. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Demonstrate that each required PORV is OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once each REFUELING INTERVAL; and

c. Verifying the PORV block valve is open and the PORV Cold Overpressure Protection System (COPPS) is armed at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Demonstrate that each required RHR suction relief valve is OPERABLE by:

a. Verifying the isolation valves between the RCS and each required RHR suction relief valve are open at least once per 12 hours; and

b. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 When complying with 3.4.9.3.4, verify that the RCS is vented through a vent pathway ≥ ~~5.4~~ square inches at least once per 31 days for a passive vent path and at least once per 12 hours for unlocked open vent valves.
2.0

4.4.9.3.4 Verify that no Safety Injection pumps are capable of injecting into the RCS at least once per 12 hours.

4.4.9.3.5 Verify that a maximum of one centrifugal charging pump is capable of injecting into the RCS at least once per 12 hours.

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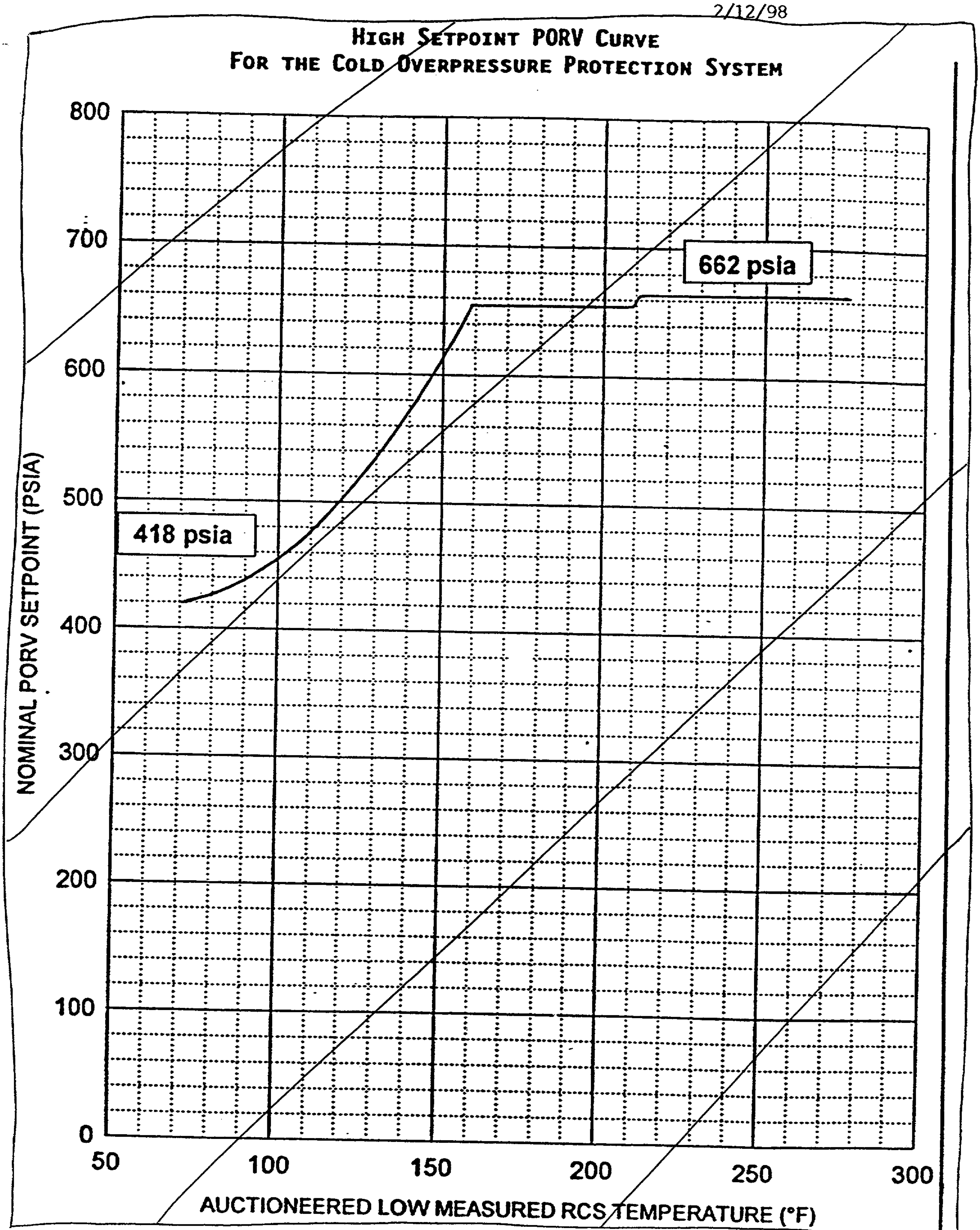
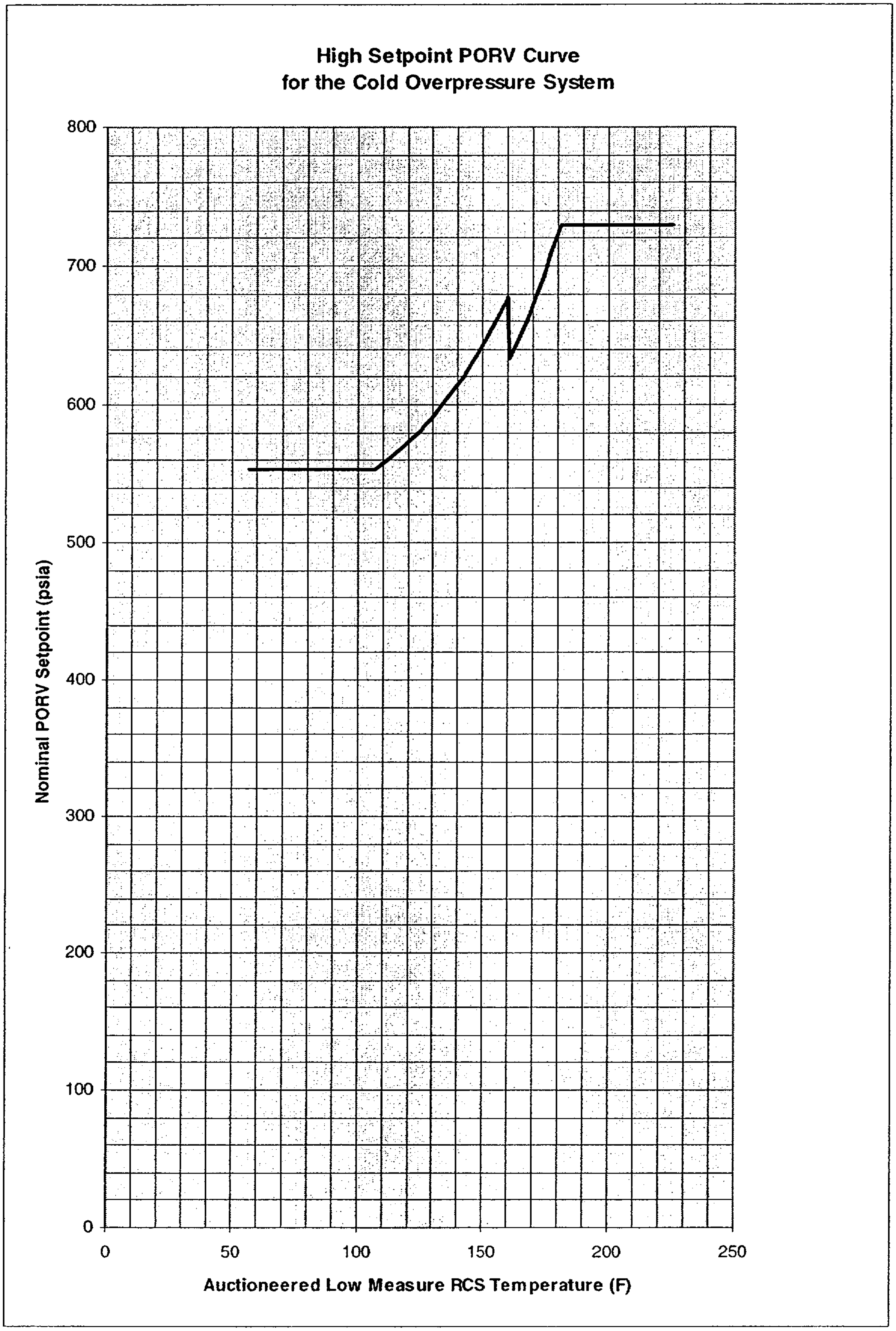


FIGURE 3.4-4a

INSERT New Figure High Setpoint


High Setpoint PORV Curve
for the Cold Overpressure System
Nominal PORV Setpoint (psia)
800
700
600
500
400
300
200
100
0
0
50
100
150
200
250
Auctioneered Low Measure RCS Temperature (F)

2/12/98

Low Setpoint PORV Curve For the Cold Overpressure Protection System

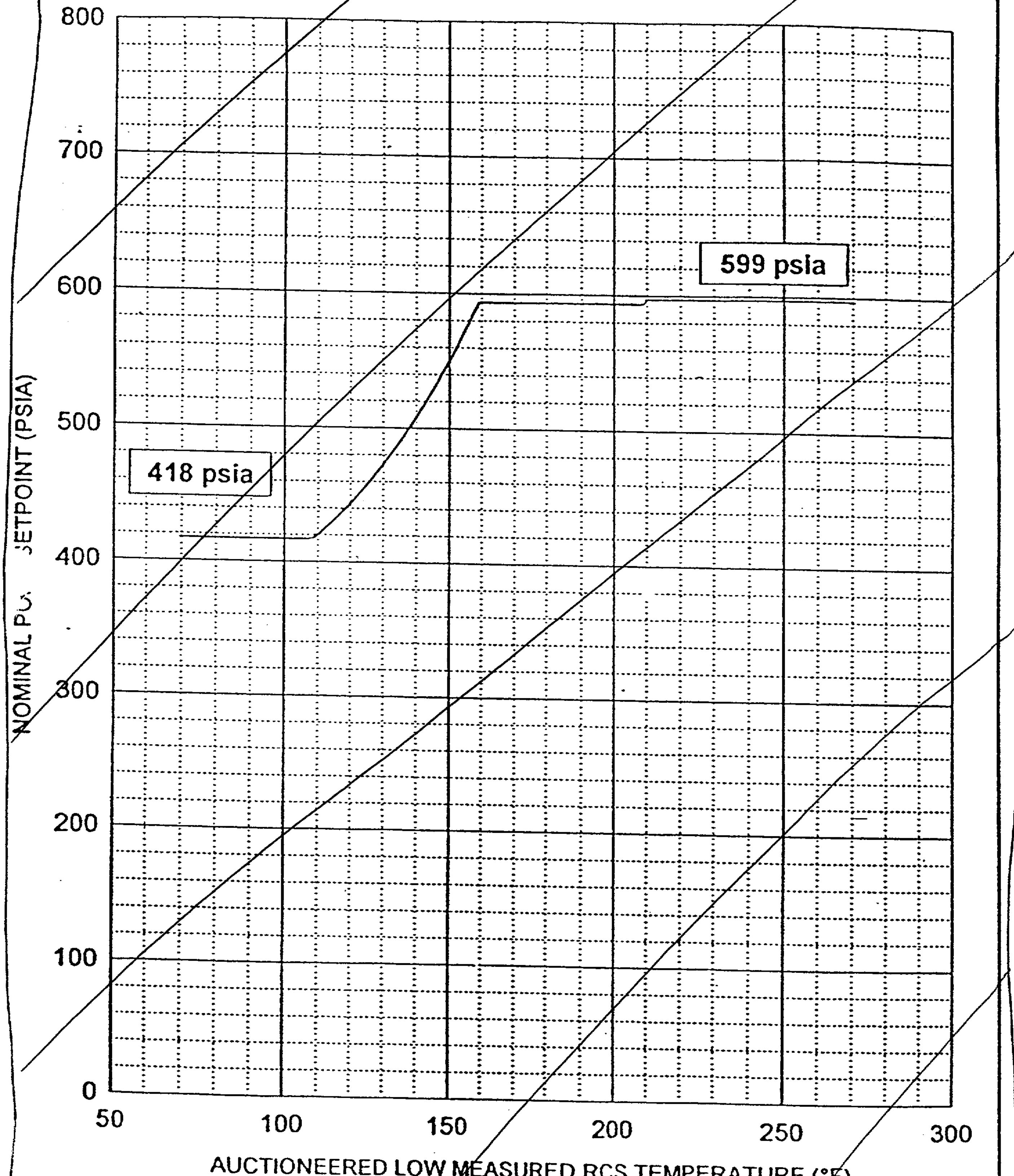
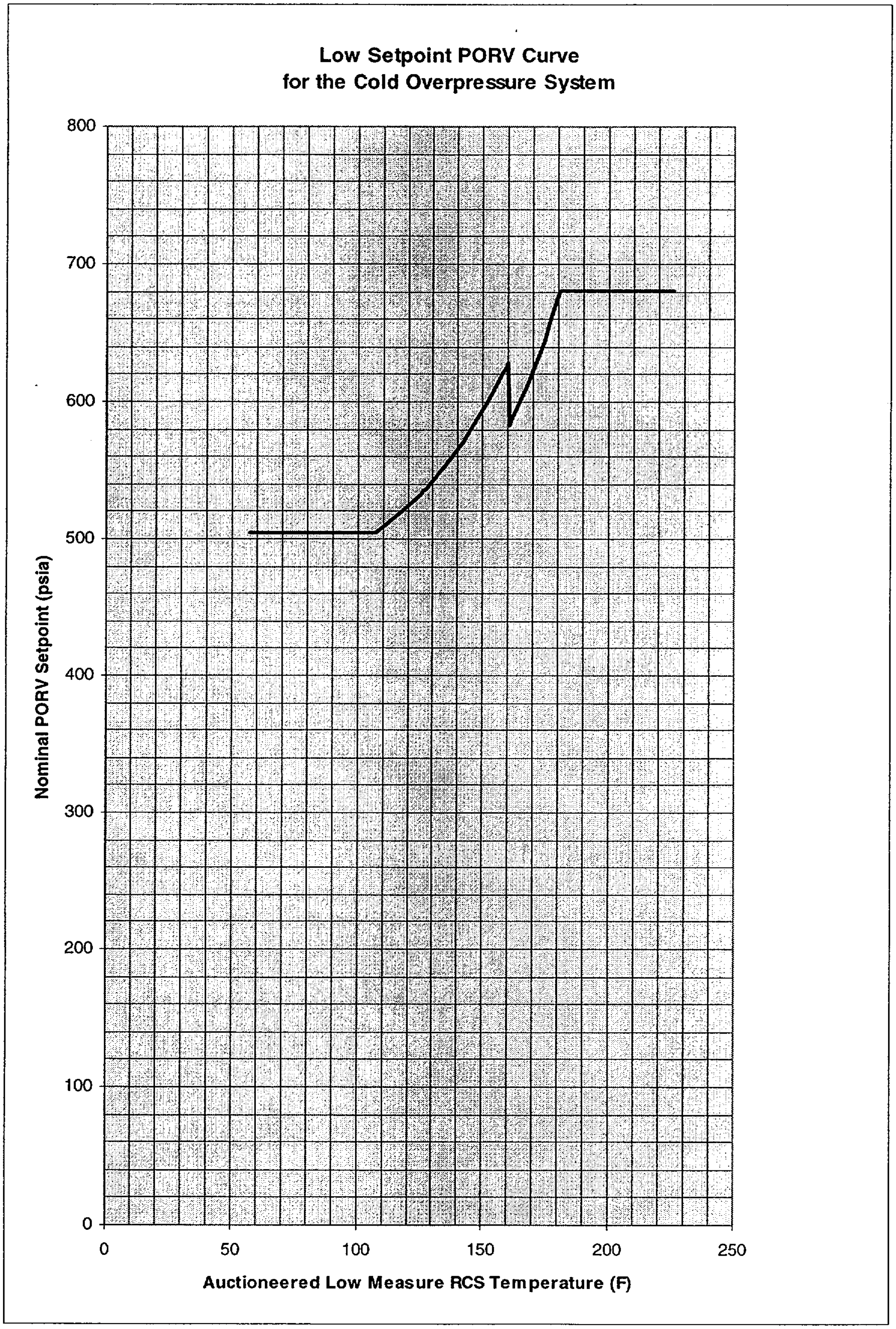


FIGURE 3.4-4b

INSERT New Figure Low Setpoint


Low Setpoint PORV Curve
for the Cold Overpressure System
800
700
600
500
400
300
200
100
0
Nominal PORV Setpoint (psia)
0
50
100
150
200
250
Auctioneered Low Measure RCS Temperature (F)

NO CHANGE
FOR INFORMATION
ONLY

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2, the most restrictive condition occurs at EOL with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta K/K$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4 and 5, the most restrictive condition occurs at BOL, associated with a boron dilution accident. In the analysis of this accident, a minimum SHUTDOWN MARGIN as defined in Specification 3/4.1.1.1.2 is required to allow the operator 15 minutes from the initiation of the Shutdown Margin Monitor alarm to total loss of SHUTDOWN MARGIN. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting requirement and is consistent with the accident analysis assumption. The required SHUTDOWN MARGIN is plotted as a function of RCS critical boron concentration.

The locking closed of the required valves in MODE 5 (with the loops not filled) will preclude the possibility of uncontrolled boron dilution of the Reactor Coolant System by preventing flow of unborated water to the RCS.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

These corrections involved: (1) a conversion of the MDC used in the FSAR safety analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR safety analyses into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative MTC value at a core condition of 300 ppm equilibrium boron concentration, and is obtained by making corrections for burnup and soluble boron to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the P-12 interlock is above its setpoint, (4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (5) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 ~~BORATION SYSTEMS~~ DELETED

~~The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.~~

~~With the plant in MODES 1, 2, or 3, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN~~

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions equivalent to that required by Figure 3.1-4 after xenon decay and cooldown to 200°F. The maximum boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The initial RCS boron concentration is based on a minimum expected hot full power or hot zero power condition (peak xenon). The final RCS boron concentration assumes that the most reactive control rod is not inserted into the core. This set of conditions requires a minimum usable volume of 21,802 gallons of 6600 ppm borated water from the boric acid storage tanks or 1,166,000 gallons of 2700 ppm borated water from the refueling water storage tank (RWST). A minimum RWST volume of 1,166,000 gallons is specified to be consistent with ECCS requirement.

With the plant in MODE 4, one boron injection flowpath is acceptable without single failure consideration for emergency boration requirements on the basis of the stable reactivity condition of the reactor, the emergency power supply requirement for the OPERABLE charging pump, and the fact that the plant is administratively borated to at least MODE 5 requirements prior to cooldown to MODE 4. Also, the primary grade water addition path to the charging pumps is surveilled to be locked closed to prevent a direct dilution accident in MODE 4.

With the plant in MODES 5 and 6, one boron injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection system becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE, when cold overpressure protection is in service, provides assurance that a mass addition pressure transient can be relieved by operation of a single PORV or RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.3% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either a usable volume of 4100 gallons of 6600 ppm borated water from the boric acid storage tanks or 250,000 gallons of 2700 ppm borated water from the RWST. The unusable volume in each boric acid storage tank is 1300 gallons.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The minimum RWST solution temperature for MODES 5 and 6 is based on analysis assumptions in addition to freeze protection considerations. The minimum/maximum RWST solution temperatures for MODES 1, 2, 3 and 4 are based on analysis assumptions.

REACTIVITY CONTROL SYSTEMS

BASES

~~BORATION SYSTEMS (Continued)~~

~~The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.~~

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate in MODES 1 and 2 with three or four reactor coolant loops in operation and maintain DNBR greater than the design limit during all normal operations and anticipated transients. With less than the required reactor coolant loops in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, three reactor coolant loops, and in Mode 4, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, in MODE 3 a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., ~~by opening the Reactor Trip System breakers~~. the Control Rod Drive System is not capable of rod withdrawal

In MODE 4, if a bank withdrawal accident can be prevented, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (any combination of RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two RHR loops or at least one RHR loop and two steam generators be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

In MODE 5, during a planned heatup to MODE 4 with all RHR loops removed from operation, an RCS loop, OPERABLE and in operation, meets the requirements of an OPERABLE and operating RHR loop to circulate reactor coolant. During the heatup there is no requirement for heat removal capability so the OPERABLE and operating RCS loop meets all of the required functions for the heatup condition. Since failure of the RCS loop, which is OPERABLE and operating, could also cause the associated steam generator to be inoperable, the associated steam generator cannot be used as one of the steam generators used to meet the requirement of LCO 3.4.1.4.1.b.

INSERT H

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

~~The restrictions on starting an RCP are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50 or which could cause pressure excursions within the RHR system which would exceed the design pressure of the system. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs based upon the secondary water temperature of each steam generator and the RCS wide-range cold leg temperatures.~~

INSERT H - Page B 3/4 4-1

The restrictions on starting the first RCP in MODE 4 below the cold overpressure protection enable temperature (226 °F), and in MODE 5 are provided to prevent RCS pressure transients. These transients, energy additions due to the differential temperature between the steam generator secondary side and the RCS, can result in pressure excursions which could challenge the P/T limits. The RCS will be protected against overpressure transients and will not exceed the reactor vessel isothermal beltline P/T limit by restricting RCP starts based on the differential water temperature between the secondary side of each steam generator and the RCS cold legs. The restrictions on starting the first RCP only apply to RCPs in RCS loops that are not isolated. The restoration of isolated RCS loops is normally accomplished with all RCPs secured. If an isolated RCS loop is to be restored when an RCP is operating, the appropriate temperature differential limit between the secondary side of the isolated loop steam generator and the in service RCS cold legs is applicable, and shall be met prior to opening the loop isolation valves.

The temperature differential limit between the secondary side of the steam generators and the RCS cold legs is based on the equipment providing cold overpressure protection as required by Technical Specification 3.4.9.3. If the pressurizer PORVs are providing cold overpressure protection, the steam generator secondary to RCS cold leg water temperature differential is limited to a maximum of 50 °F. If any RHR relief valve is providing cold overpressure protection and RCS cold leg temperature is above 150 °F, the steam generator secondary water temperature must be at or below RCS cold leg water temperature. If any RHR relief valve is providing cold overpressure protection and RCS cold leg temperature is at or below 150 °F, the steam generator secondary to RCS cold leg water temperature differential is limited to a maximum of 50 °F.

isolated first

The requirement to maintain the isolated loop stop valves shut with power removed ensures that no reactivity addition to the core could occur due to the startup of an isolated loop. Verification of the boron concentration in an ~~idle~~ loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. ~~The 2600 ppm is sufficient to bound shutdown margin requirements and provide for boron concentration measurement uncertainty between the loop and the RWST. Draining and refilling the isolated loop within 4 hours prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.~~

~~The requirement to have all reactor coolant pumps de-energized, prior to unisolating a loop, insures that the heat from the secondary side of the steam generator, in the loop being unisolated, does not result in an energy addition transient during the return of the loop to service.~~

INSERT I

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. ~~Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Cold Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.~~

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The pressurizer provides a point in the RCS when liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during load transients.

MODES 1 AND 2

The requirement for the pressurizer to be OPERABLE, with pressurizer level maintained at programmed level within ± 6% of full scale is consistent with the accident analysis in Chapter 15 of the FSAR. The accident analysis assumes that pressurizer level is being maintained at the programmed level by the automatic control system, and when in manual control, similar limits are established. The programmed level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure and pressurizer overfill transients. A pressurizer level control error based upon automatic level control has been taken into account for those transients where pressurizer overfill is a concern (e.g., loss of feedwater, feedwater line break, and inadvertent ECCS actuation at power). When in manual control, the goal is to maintain pressurizer level at the program level value. The ±6 % of full scale acceptance criterion in the Technical Specification establishes a band for operation to accommodate variations between level measurements. This value is bounded by the margin applied to the pressurizer overfill events.

INSERT I - Page B 3/4 4-2

If any pressurizer Code safety valve is inoperable, and cannot be restored to OPERABLE status, the action statement requires the plant to be shut down and cooled down such that Technical Specification 3.4.9.3 will be come applicable and require cold overpressure protection to be placed in service.

SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM (EXCEPT THE PRESSURIZER)

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

~~The LCO and~~ Figures 3.4-2 and 3.4-3 contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational requirements during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. A heatup or cooldown is defined as a temperature increase or decrease of greater than or equal to 10°F in any one hour period. This definition of heatup and cooldown is based upon the ASME definition of isothermal conditions described in ASME, Section XI, Appendix E.

PRESSURE/TEMPERATURE LIMITS (continued)

Steady state thermal conditions exist when temperature increases or decreases are <10°F in any one hour period and when the plant is not performing a planned heatup or cooldown in accordance with a procedure. During steady state thermal conditions, the limits of the heatup and cooldown curves do not apply. Cold overpressure protection is adequate to protect the reactor coolant system.

applicable to the ferritic material

The LCO establishes operating limits that provide a margin to brittle failure ~~of the reactor vessel and piping~~ of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the Pressurizer, which has different design characteristics and operating functions which are addressed by LCO 3.4.9.2, "Pressurizer".

INSERT J

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

4

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. ~~3~~) and Appendix H of 10 CFR 50 (Ref. ~~4~~). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. ~~5~~). 5

6

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations may be more restrictive, and thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The P/T limits include uncertainty margins to ensure that the calculated limits are not inadvertently exceeded. These margins include gauge and system loop uncertainties, elevation differences, containment pressure conditions and system pressure drops between the beltline region of the vessel and the pressure gauge or relief valve location. In an effort to minimize the system frictional losses, additional restrictions on RCP operation below 160°F are provided in the LCO. These restrictions result in increased acceptable system pressures enabling greater operator flexibility during heatup and cooldown in MODE 5.

INSERT J - Page B 3/4 4-8

The P/T limits have been established for the ferritic materials of the RCS considering ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 1) as modified by ASME Code Case N-640 (Reference 2), and the additional requirements of 10 CFR 50 Appendix G (Reference 3). Implementation of the specific requirements provide adequate margin to brittle fracture of ferritic materials during normal operation, anticipated operational occurrences, and system leak and hydrostatic tests.

PRESSURE/TEMPERATURE LIMITS (continued)

This limit provides the required margin relative to brittle fracture.

160°F above

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.4, "Minimum Temperature for Criticality."

ferritic RCPB materials

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the ~~RCPB~~, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. ~~6~~ 7) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSIS

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. ~~Reference 2 establishes~~ the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

INSERT K

RCS P/T limits satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

~~The two elements of this LCO are:~~

~~a. The limit curves for heatup, cooldown, and ISLH testing; and~~

~~b. Limits on the rate of change of temperature.~~

the ferritic

The LCO limits apply to ~~all~~ components of the RCS, except the Pressurizer. These limits define allowable operating regions while providing margin against nonductile failure for the controlling ferritic component.

~~The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curve. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.~~

INSERT L

<u>INSERT K - Page B 3/4 4-9</u>

Reference 1, as modified by Reference 2, combined with the additional requirements of Reference 3 provide

<u>INSERT L - Page B 3/4 4-9</u>

The limitations imposed on the rate of change of temperature have been established to ensure consistency with the resultant heatup, cooldown, and ISLH testing P/T limit curves. These limits control the thermal gradients (stresses) within the reactor vessel beltline (the limiting component). Note that while these limits are to provide protection to ferritic components within the reactor coolant pressure boundary, a limit of 100°F/hr applies to the reactor coolant pressure boundary (except the pressurizer) to ensure that operation is maintained within the ASME Section III design loadings, stresses, and fatigue analyses for heatup and cooldown.

PRESSURE/TEMPERATURE LIMITS (continued)

Violating the LCO limits places the reactor vessel outside of the bounds of the analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

of ferritic RCS components using ASME Section XI, Appendix G, as modified by Code Case N-640 and the additional requirements of

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure ~~in accordance with~~ 10CFR50, Appendix G (Ref. 1). The P/T limits were developed to provide requirements for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, in keeping with the concern for nonductile failure. The limits do not apply to the Pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.2.5, "DNB Parameters"; LCO 3.2.3.1 and 3.2.3.2, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - Four Loops Operating/Three Loops Operating"; LCO 3.1.1.4, "Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS 5

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The ~~30 minute~~ Allowed Outage Time (AOT) reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

PRESSURE/TEMPERATURE LIMITS (continued)

7

ASME Code, Section XI, Appendix E (Ref. ~~6~~), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

When operating in MODES 1 through 4

The 72 hour AOT is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

the AOT

This evaluation must be completed whenever a limit is exceeded. Restoration within ~~30 minutes~~ alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity. or not allowed to enter MODE 4

If the required remedial actions are not completed within the allowed times, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

~~If the required restoration activity cannot be accomplished within 30 minutes, action must be implemented to reduce pressure and temperature as specified in the ACTION statement.~~

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in the Action statement. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. IN MODES 1 through 4

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psia within the next 30 hours.

The AOTs are reasonable, based on operating experience to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

INSERT M

SURVEILLANCE REQUIREMENTS

Verification that operation is within the LCO limits as well as the limits of Figures 3.4-2 and 3.4-3 is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This frequency is considered reasonable in view of the control room indication available to monitor RCS status.

INSERT M - Page B 3/4 4-11

Completion of the required evaluation following limit violation in other than MODES 1 through 4 is required before plant startup to MODE 4 can proceed.

PRESSURE/TEMPERATURE LIMITS (continued)

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This Surveillance Requirement is only required to be performed during system heatup, cooldown, and ISLH testing. No Surveillance Requirement is given for criticality operations because LCO 3.1.1.4 contains a more restrictive requirement.

The Surveillance Requirement to remove and examine the reactor vessel material irradiation surveillance specimens is in accordance with the requirements of 10CFR50, Appendix H.

INSERT N

REFERENCES

1. ~~10CFR50, Appendix G.~~
2. ~~ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.~~
3. ~~ASTM E 185-82, July 1982.~~
4. ~~10CFR50, Appendix H.~~
5. ~~Regulatory Guide 1.99, Revision 2, May 1988.~~
6. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.~~

INSERT O

PRESSURIZER

BACKGROUND

The Pressurizer is part of the RCPB, but is not subject to the same restrictions as the rest of the RCS. This LCO limits the temperature changes of the Pressurizer and allowable temperature differentials, within the design assumptions and the stress limits for cyclic operation.

INSERT N - Page B 3/4 4-12

It is not necessary to perform Surveillance Requirement 4.4.9.1.1 to verify compliance with Figures 3.4-2 and 3.4-3 when the reactor vessel is fully detensioned. During refueling, with the head fully detensioned or off the reactor vessel, the RCS is not capable of being pressurized to any significant value. The limiting thermal stresses which could be encountered during this time would be limited to flood-up using RWST water as low as 40°F. It is not possible to cause crack growth of postulated flaws in the reactor vessel at normal refueling temperatures even injecting 40°F water.

INSERT O - Page B 3/4 4-12

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.

2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.

3. 10 CFR 50 Appendix G, "Fracture Toughness Requirements."

4. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706."

5. 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

6. Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events," 1995 Edition.

PRESSURIZER (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure as specified in the Action statement. A favorable evaluation must be completed and documented before returning to operating pressure conditions.

Pressure is reduced by bringing the plant to MODE 3 within 6 hours. Pressure is further reduced by bringing the plant to MODE 4 or 5 and reducing Pressurizer pressure < 500 psia within the next 30 hours.

The AOTs are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Verification that operation is within the LCO heatup and cooldown limits is required every 30 minutes when Pressurizer temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor Pressurizer status. Surveillance for heatup or cooldown may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied. The Surveillance Requirement for heatup or cooldown is only required to be performed during system heatup and cooldown.

Verification that operation is within the LCO spray water temperature differential limit is required every 12 hours when auxiliary spray is in operation. This Frequency is considered reasonable in view of the control room indication available to monitor Pressurizer status.

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

Isothermal beltline

INSERT P

The Cold Overpressure Protection System limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits ~~of 10CFR50, Appendix G (Ref. 1).~~ The reactor vessel is the limiting RCPB component for demonstrating such protection.

Cold Overpressure Protection consists of two PORVs with nominal lift setting as specified in Figures 3.4-4a and 3.4-4b, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two relief valves are required for redundancy. One relief valve has adequate relieving capability to prevent overpressurization of the RCS for the required mass input capability.

INSERT P - Page B 3/4 4-15

developed using the guidance of ASME Section XI, Appendix G (Reference 1) as modified by ASME Code Case N-640 (Reference 2).

OVERPRESSURE PROTECTION SYSTEMS (continued)

The use of a PORV for Cold Overpressure Protection is limited to those conditions when no more than one RCS loop is isolated from the reactor vessel ~~and, whenever an RCP is running, the temperature signal from the isolated loop is removed from the PORV opening logic~~. When two or more loops are isolated, Cold Overpressure Protection must be provided by either the two RHR suction relief valves or a depressurized and vented RCS.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to stress at low temperatures (Ref. ~~2~~ 3). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause nonductile cracking of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits - Reactor Coolant System," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the limits provided in Figures 3.4-2 and 3.4-3.

This LCO provides RCS overpressure protection by limiting mass input capability and requiring adequate pressure relief capacity. Limiting mass input capability requires all Safety Injection (SIH) pumps and all but one centrifugal charging pump to be incapable of injection into the RCS. The pressure relief capacity requires either two redundant relief valves or a depressurized RCS and an RCS vent of sufficient size. One relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum mass input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the Cold Overpressure Protection MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve.

INSERT Q

PORV Requirements

As designed, the PORV Cold Overpressure Protection (COPPS) is signaled to open if the RCS pressure approaches a limit determined by the COPPS actuation logic. The COPPS actuation logic monitors both RCS temperature and RCS pressure and determines when the nominal setpoint of Figure 3.4-4a or Figure 3.4-4b is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

INSERT Q - Page B 3/4 4-16

If a loss of RCS inventory or reduction in shutdown margin event occurs, the appropriate response will be to correct the situation by starting RCS makeup pumps. If the loss of inventory or shutdown margin is significant, this may necessitate the use of additional RCS makeup pumps that are being maintained not capable of injecting into the RCS in accordance with Technical Specification 3.4.9.3. The use of these additional pumps to restore RCS inventory or shutdown margin will require entry into the associated action statement. The action statement requires immediate action to comply with the specification. The restoration of RCS inventory or shutdown margin can be considered to be part of the immediate action to restore the additional RCS makeup pumps to a not capable of injecting status. While recovering RCS inventory or shutdown margin, RCS pressure will be maintained below the P/T limits. After RCS inventory or shutdown margin has been restored, the additional pumps should be immediately made not capable of injecting and the action statement exited.

OVERPRESSURE PROTECTION SYSTEMS (continued)

PORV Requirements (continued)

measured

measured by

The lowest temperature signal is processed through a function generator that calculates a pressure setpoint for that temperature. The calculated pressure setpoint is then compared with ~~the indicated~~ RCS pressure ~~from~~ a wide range pressure channel. If the ~~indicated~~ pressure meets or exceeds the calculated value, a PORV is signaled to open.

~~The wide range RCS temperature indicators monitor loop temperature in the portion of the RCS loop that can be isolated. If a loop is isolated, the temperature signal from the isolated loop may be significantly lower than the temperature signals from the unisolated loops. This would result in a calculated pressure setpoint for the PORV below that anticipated by the operator based on the temperature in the unisolated portion of the RCS. Since this could result in a significantly lower calculated PORV setpoint, RCP operation is not permitted under these conditions unless the temperature input from the isolated loop is removed from the auctioneered circuit. This restriction will ensure that the #1 RCP seals are not challenged as a result of PORV undershoot. Since~~ the PORV mass and heat injection transients have only been analyzed for a maximum of one loop isolated, the use of the PORVs is restricted to three and four RCS loops unisolated.

If ~~one loop is isolated without removing its temperature input from the PORV calculated setpoint auctioneered circuit and at least one RCP is in operation, or if~~ more than one loop is isolated, then the PORVs must have their block valves closed or COPPS must be blocked. For these cases, Cold Overpressure Protection must be provided by either the two RHR suction relief valves or a depressurized RCS and an RCS vent.

This is necessary because

The use of the PORVs is restricted to three and four RCS loops unisolated; for a loop to be considered isolated, both RCS loop stop valves must be closed. ~~If only one loop stop valve in an RCS loop is closed for an extended period, the PORVs must have their block valves closed or COPPS must be blocked. A single RCS loop stop valve can be stroked for short time periods for surveillances or other purposes and not affect the use of the PORVs for Cold Overpressure Protection.~~

The RHR suction relief valves have been qualified for all mass injection transients for any combination of isolated loops. In addition, the heat injection transients not prohibited by the Technical Specifications have also been considered in the qualification of the RHR suction relief valves.

Figure 3.4-4a and Figure 3.4-4b present the PORV setpoints for COPPS. ~~Above 110°F, the~~ setpoints are staggered so only one valve opens during a low

The

OVERPRESSURE PROTECTION SYSTEMS (continued)

temperature overpressure transient. Setting both valves to the values of Figure 3.4-4a and Figure 3.4-4b within the tolerance allowed for the calibration accuracy, ensures that the ~~Reference 1~~ isothermal P/T limits will not be exceeded for the analyzed isothermal events.

When a PORV is opened, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RHR Suction Relief Valve Requirements

The isolation valves between the RCS and the RHR suction relief valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows type water relief valves with setpoint tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. ~~3~~ 4) for Class 2 relief valves.

When the RHR system is operated for decay heat removal or low pressure letdown control, the isolation valves between the RCS and the RHR suction relief valves are open, and the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at ~~containment ambient pressure~~ acceptable pressure levels in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting mass or heat input transient, and maintaining pressure below the P/T limits for the analyzed isothermal events.

For an RCS vent to meet the flow capacity requirement, it requires removing a Pressurizer safety valve, removing a Pressurizer manway, or similarly establishing a vent by opening an RCS vent valve provided that the opening meets the ~~size~~ relieving capacity requirements. The vent path must be above the level of reactor coolant, so as not to drain the RCS when open.

OVERPRESSURE PROTECTION SYSTEMS (continued)

APPLICABLE SAFETY ANALYSIS

Safety analyses (Ref. ~~4~~ 5) demonstrate that the reactor vessel is adequately protected against exceeding the P/T limits for the analyzed isothermal events. In MODES 1, 2, AND 3, and in MODE 4, with RCS cold leg temperature exceeding ~~275~~ 226°F, the pressurizer safety valves will provide RCS overpressure protection in the ductile region. At ~~275~~ 226°F and below, overpressure prevention is provided by two means: (1) two OPERABLE relief valves, or (2) a depressurized RCS with a sufficiently sized RCS vent, ~~as required by NUREG-0800, RSB 5-2,~~ for temperatures less than RT_{NDT} + ~~90~~ 50°F. Each of these means has a limited overpressure relief capability. Consistent with ASME Section XI, Appendix G

The required RCS temperature for a given pressure increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the Technical Specification curves are revised, the cold overpressure protection must be re-evaluated to ensure its functional requirements continue to be met using the RCS relief valve method or the depressurized and vented RCS condition.

Transients capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch

Heat Input Transients

a. Inadvertent actuation of Pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The Technical Specifications ensure that mass input transients beyond the operability of the cold overpressure protection means do not occur by rendering all Safety Injection Pumps and all but one centrifugal charging pump incapable of injecting into the RCS whenever ~~an RHR suction relief valve is unisolated from the RCS or whenever any PORV has COPPS armed and its block valve open.~~ any RCS cold leg is ≤ 226°F.

The Technical Specifications ensure that energy addition transients beyond the operability of the cold overpressure protection means do not occur by limiting reactor coolant pump starts. LCO 3.4.1.4.1, "Reactor Coolant Loops and Coolant Circulation - Cold Shutdown - Loops Filled," LCO 3.4.1.4.2, "Reactor Coolant

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OVERPRESSURE PROTECTION SYSTEMS (continued)

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Loops and Coolant Circulation - Cold Shutdown - Loops Not Filled," and LCO 3.4.1.3, "Reactor Coolant Loops and Coolant Circulation - Hot Shutdown" limit reactor coolant pump starts to one of the following plant conditions:

a. An RCP is running, and
The wide range cold leg temperature of any unisolated RCS loop is >160°F, or

b. Two or more RCS loops are isolated, and
An RCP is not running, and
The secondary side water temperature of any steam generator in an unisolated loop is equal to or less than the wide range cold leg temperature of any unisolated RCS loop, or

c. No more than one RCS loop is isolated, and
An RCP is not running, and
Any RHR suction relief valve is unisolated from the RCS, and
The secondary side water temperature of any steam generator in an unisolated loop is either:

- >200°F and equal to or less than the wide range cold leg temperature of any unisolated RCS loop, or
- ≤200°F and ≤50°F hotter than the wide range cold leg temperature of any unisolated RCS loop. (Note: Reactor coolant pumps cannot be run with the wide range cold leg temperature of any unisolated RCS loop <160°F if any PORV has COPPS armed and has its block valve open.), or

d. No more than one RCS loop is isolated, and
An RCP is not running, and
The RHR suction relief valves are isolated from the RCS, and
The wide range cold leg temperature of any unisolated RCS loop ≥160°F, and
Any PORV has COPPS armed and has its block valve open, and
The secondary side water temperature of any steam generator in an unisolated loop is either:

- equal to or less than the wide range cold leg temperature of any unisolated RCS loop, or
- <250°F and ≤50°F hotter than the wide range cold leg temperature of any unisolated RCS loop,

or

e. The RHR suction relief valves are isolated from the RCS, and
Both PORVs are isolated or COPPS is blocked, and
The wide range cold leg temperature of any unisolated RCS loop is >275°F.

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starting the first reactor coolant pump such that it shall not be started when any RCS loop wide range cold leg temperature is $\leq$ 226 °F unless the secondary side water temperature of each steam generator is < 50 °F above each RCS cold leg temperature. The restrictions ensure the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressurization event beyond the capability of the COPPS to mitigate. The COPPS utilizes the pressurizer PORVs and the RHR relief valves to mitigate the limiting mass and energy addition events, thereby protecting the isothermal reactor vessel beltline P/T limits. The restrictions will ensure the reactor vessel will be protected from a cold overpressure event when starting the first RCP. If at least one RCP is operating, no restrictions are necessary to start additional RCPs for reactor vessel protection. In addition, this restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

The RCP starting criteria are based on the equipment used to provide cold overpressure protection. A maximum temperature differential of 50 °F between the steam generator secondary sides and RCS cold legs will limit the potential energy addition to within the capability of the pressurizer PORVs to mitigate the transient. The RHR relief valves are also adequate to mitigate energy addition transients constrained by this temperature differential limit, provided all RCS cold leg temperatures are at or below 150 °F. The ability of the RHR relief valves to mitigate energy addition transients when RCS cold leg temperature is above 150 °F has not been analyzed. As a result, the temperature of the steam generator secondary sides must be at or below the RCS cold leg temperatures if the RHR relief valves are providing cold overpressure protection and the RCS cold leg temperature is above 150 °F

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OVERPRESSURE PROTECTION SYSTEMS (continued)

The cold overpressure transient analyses demonstrate that either one relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when RCS letdown is isolated and only one centrifugal charging pump is operating. Thus, the LCO allows only one centrifugal charging pump capable of injecting when cold overpressure protection is required.

The cold overpressure protection enabling temperature is conservatively established at a value ≤ 226°F ~~≥ 275°F~~ based on the criteria provided by ASME Section XI, Appendix G. ~~described in Branch Technical Position RSB 5-2 provided in the Standard Review Plan (NUREG-0800).~~

PORV Performance

The ~~10CFR50 Appendix G~~ analyses show that the vessel is protected against non-ductile failure when the PORVs are set to open at the values shown in Figures 3.4-4a and 3.4-4b within the tolerance allowed for the calibration accuracy. The curves are derived by analyses for both three and four RCS loops unisolated that model the performance of the PORV cold overpressure protection system (COPPS), assuming the limiting mass and heat transients of one centrifugal charging pump injecting into the RCS, or the energy addition as a result of starting an RCP with temperature asymmetry between the RCS and the steam generators. These analyses consider pressure overshoot ~~and undershoot~~ beyond the PORV opening setpoint ~~and closing,~~ resulting from signal processing and valve stroke times.

The PORV setpoints in Figures 3.4-4a and 3.4-4b will be updated when the P/T limits conflict with the cold overpressure analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement. Revised limits are determined using neutron fluence projections and the results of testing of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.9.1, "Pressure/Temperature Limits - Reactor Coolant System (Except the Pressurizer)," discuss these evaluations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints as do the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 426.8 psig and 453.2 psig will pass flow greater than that required for the limiting cold overpressure transient while maintaining RCS pressure less than the isothermal P/T limit curve. Assuming maximum relief flow requirements during the limiting cold overpressure event, an RHR suction relief valve will maintain RCS pressure to ≤ 110% of the nominal lift setpoint.

Although each RHR suction relief valve is a passive spring loaded device, which meets single failure criteria, its location within the RHR System precludes meeting single failure criteria when spurious RHR suction isolation valve or RHR suction valve closure is postulated. Thus the loss of an RHR suction relief

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OVERPRESSURE PROTECTION SYSTEMS (continued)

valve is the worst case single failure. Also, as the RCS P/T limits are revised to reflect change in toughness in the reactor vessel materials, the RHR suction relief valve's analyses must be re-evaluated to ensure continued accommodation of the design bases cold overpressure transients.

RCS Vent Performance

With the RCS depressurized, limiting analyses show a vent size of ≥ ~~5.4~~ 2.0 square inches is capable of mitigating the ~~allowed~~ cold overpressure transient. The capacity of this vent size is greater than the flow of the limiting transient, while maintaining RCS pressure less than the maximum pressure on the isothermal P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the isothermal P/T limit curves are revised.

The RCS vent is a passive device and is not subject to active failure.

The RCS vent satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

RCP Seal Protection

As described above, the analyses of the cold overpressure transients result in pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The valve overshoots are considered in the generation of the PORV setpoints presented in Figures 3.4-4a and 3.4-4b.

The valve undershoots are also evaluated in terms of potential damage to the RCP #1 seal. The minimum pressure, considering valve undershoot, must be higher than that required to maintain the RCP #1 seal as a film riding seal. This requirement resulted in restrictions on the operation of pumps when the cold overpressure protection is being provided by one or two PORVs. Specifically,

a. When the RCS cold leg temperature of any unisolated loop is less than 160 degrees F, the PORV block valves are open, and the PORV's Cold Overpressure Protection System (COPPS) is armed, no RCPs may be in operation. LCO 3.4.1.4.1, "Reactor Coolant Loops and Coolant Circulation - Cold Shutdown - Loops Filled," and LCO 3.4.1.4.2, "Reactor Coolant Loops and Coolant Circulation - Cold Shutdown - Loops Not Filled," provide this protection.

b. When COPPS is armed, with the steam generator secondary side ≥ 250°F, heat injection transients due to the start of the first RCP with a temperature asymmetry between the RCS and the steam generators is prohibited. LCO 3.4.1.3, "Reactor Coolant System - Hot Shutdown," provides this protection for MODE 4.

OVERPRESSURE PROTECTION SYSTEMS (continued)

c. When COPPS is armed, PORV undershoot is analyzed for mass injection transients limited to one charging pump. LCO 3.4.9.3, "Reactor Coolant System - Overpressure Protection Systems," provides this protection by requiring both safety injection pumps and all but one charging pump to be incapable of injection into the RCS.

In order to provide protection for the RCP #1 seal, a PORV setpoint of ≥ 595 psia for temperatures ≥ 160 degrees F must be met. This minimum setpoint is derived by adding the applicable train uncertainty and valve undershoot to the required minimum RCS pressure required for seal integrity. Due to the differing instrument uncertainties for the two trains of PORV COPPS, the train with the highest uncertainty is paired to the high setpoint curve.

LCO

This LCO requires that cold overpressure protection be OPERABLE and the maximum mass input be limited to one charging pump. Failure to meet this LCO could lead to the loss of low temperature overpressure mitigation and violation of the reactor vessel ~~Reference 1~~ isothermal P/T limits as a result of an operational transient.

To limit the mass input capability, the LCO requires a maximum of one centrifugal charging pump capable of injecting into the RCS.

The elements of the LCO that provides low temperature overpressure mitigation through pressure relief are:

1. Two OPERABLE PORVs; or

   A PORV is OPERABLE for cold overpressure protection when its block valve is open, its lift setpoint is set to the nominal setpoints provided for both three and four loops unisolated by Figure 3.4-4a or 3.4-4b and when the surveillance requirements are met. ~~For three loops unisolated, the temperature input from the isolated loop must be removed from the COPPS auctioneered circuitry whenever any RCP is in operation.~~

2. Two OPERABLE RHR suction relief valves; or

   An RHR suction relief valve is OPERABLE for cold overpressure protection when its isolation valves from the RCS are open and when its setpoint is at or between 426.8 psig and 453.2 psig, as verified by required testing.

3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

4. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ ~~5.4~~ 2.0 square inches.

Each of these methods of ovepressure prevention is capable of mitigating the limiting cold overpressure transient.

OVERPRESSURE PROTECTION SYSTEMS (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ ~~275~~ 226°F, in MODE 5, and in MODE 6 when the head is on the reactor vessel. The Pressurizer safety valves provide RCS overpressure protection in the ductile region (i.e. > ~~275~~ 226°F). When the reactor head is off, overpressurization cannot occur.

LCO 3.4.9.1 "Pressure/Temperature Limits" provides the operational P/T limits for all MODES. LCO 3.4.2.~~1~~, "Safety Valves ~~- Operating~~," requires the OPERABILITY of the Pressurizer safety valves that provide overpressure protection during MODES 1, 2, ~~and~~ 3, and 4 when all RCS cold leg temperatures are > 226°F. ~~and LCO 3.4.2.1, "Safety Valves - Shutdown," requires the OPERABILITY of the Pressurizer safety valves that provide overpressure protection during MODE 4.~~

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a rapid increase in RCS pressure when little or no time exists for operator action to mitigate the event.

ACTIONS

a. and b.

With two or more centrifugal charging pumps capable of injecting into the RCS, or with any SIH pump capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted mass input capability to the RCS reflects the urgency of removing the RCS from this condition.

Required Action a. is modified by a Note that permits two centrifugal charging pumps capable of RCS injection for ≤ 1 hour to allow for pump swaps. This is a controlled evolution of short duration and the procedure prevents having two charging pumps simultaneously out of pull-to-lock while both charging pumps are capable of injecting into the RCS.

c.

In MODE 4 when any RCS cold leg temperature is ≤ ~~275~~ 226°F, with one required relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within an allowed outage time (AOT) of 7 days. Two relief valves in any combination of the PORVs and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

OVERPRESSURE PROTECTION SYSTEMS (continued)

The AOT in MODE 4 considers the facts that only one of the relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. The RCS must be depressurized and a vent must be established within the following 12 ~~8~~ hours if the required relief valve is not restored to OPERABLE within the required AOT of 7 days.

d.

The consequences of operational events that will ~~overpressure~~ overpressurize the RCS are more severe at lower temperatures (Ref. ~~7~~ 8). Thus, with one of the two required relief valves inoperable in MODE 5 or in MODE 6 with the head on, the AOT to restore two valves to OPERABLE status is 24 hours.

The AOT represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE relief valve to protect against overpressure events. The RCS must be depressurized and a vent must be established within the following ~~8~~ 12 hours if the required relief valve is not restored to OPERABLE within the required AOT of 24 hours.

e.

The RCS must be depressurized and a vent must be established within ~~8~~ 12 hours when both required Cold Overpressure Protection relief valves are inoperable.

The vent must be sized ≥ ~~5.4~~ 2.0 square inches to ensure that the flow capacity is greater than that required for the worst case cold overpressure transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible non-ductile failure of the reactor vessel.

The time required to place the plant in this Condition is based on the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1

Performance of an ANALOG CHANNEL OPERATIONAL TEST is required within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The ANALOG CHANNEL OPERATIONAL TEST will verify the setpoint in accordance with the nominal values given in Figures 3.4-4a and 3.4-4b. PORV actuation could depressurize the RCS; therefore, valve operation is not required.

OVERPRESSURE PROTECTION SYSTEMS (continued)

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required once each REFUELING INTERVAL to adjust the channel so that it responds and the valve opens within the required range and accuracy to a known input.

The PORV block valve must be verified open and COPPS must be verified armed every 72 hours to provide a flow path and a cold overpressure protection actuation circuit for each required PORV to perform its function when required. The valve is remotely verified open in the main control room. This Surveillance is performed if credit is being taken for the PORV to satisfy the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the open position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure transient.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify the PORV block valve remains open.

4.4.9.3.2

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying the RHR suction valves, 3RHS*MV8701A and 3RHS*M8701C, are open when suction relief valve 3RHS*RV8708A is being used to meet the LCO and by verifying the RHR suction valves, 3RHS*MV8702B and 3RHS*MV8702C, are open when suction relief valve 3RHS*RV8708B is being used to meet the LCO. Each required RHR suction relief valve shall also be demonstrated OPERABLE by testing it in accordance with 4.0.5. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction valves are verified to be open every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open.

The ASME Code, Section XI (Ref. ~~8~~), test per 4.0.5 verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

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OVERPRESSURE PROTECTION SYSTEMS (continued)

4.4.9.3.3

2.0

The RCS vent of ≥ ~~5.4~~ square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent valve that cannot be locked open.

b. Once every 31 days for a valve that is locked, sealed, or secured in position or any other passive vent path. A removed Pressurizer safety valve fits this category.

This passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO.

4.4.9.3.4 and 4.4.9.3.5

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SIH pumps and all but one centrifugal charging pump are verified incapable of injecting into the RCS.

The SIH pumps and charging pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. Alternate methods of control may be employed using at least two independent means to prevent an injection into the RCS. This may be accomplished through any of the following methods: 1) placing the pump in pull to lock (PTL) and pulling its UC fuses, 2) placing the pump in pull to lock (PTL) and closing the pump discharge valve(s) to the injection line, 3) closing the pump discharge valve(s) to the injection line and either removing power from the valve operator(s) or locking manual valves closed, and 4) closing the valve(s) from the injection source and either removing power from the valve operator(s) or locking manual valves closed.

An SIH pump may be energized for testing or for filling the Accumulators provided it is incapable of injecting into the RCS.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

REFERENCES

INSERT 5

~~1. 10CFR50, Appendix G~~
3. ~~2.~~ Generic Letter 88-11
4. ~~3.~~ ASME, Boiler and Pressure Vessel Code, Section III
5. ~~4.~~ FSAR, Chapter 15
6. ~~5.~~ 10CFR50, Section 50.46
7. ~~6.~~ 10CFR50, Appendix K
8. ~~7.~~ Generic Letter 90-06
9. ~~8.~~ ASME, Boiler and Pressure Vessel Code, Section XI

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1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.

2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.