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

Dominion

FEB 1 4 2002

Docket Nos. 50-336
50-423
B18556

RE: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit Nos. 2 and 3
Technical Specifications Change Requests 2-21-01 and 3-18-01
Relocation of Selected Technical Specifications Related to the Reactor Coolant System and Plant Systems

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating Licenses DPR-65 and NPF-49 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit Nos. 2 and 3, respectively. For Millstone Unit No. 2, DNC is proposing to change Technical Specifications 3/4.4.9.1, "Pressure/Temperature Limits," 3/4.7.2.1, "Steam Generator Pressure/Temperature Limitation," 3/4.7.5.1, "Flood Level," 3/4.7.7.1, "Sealed Source Contamination," and 3/4.7.8, "Snubbers." Index pages VIII and XIII will also be changed consistent with the relocation of the identified technical specifications. For Millstone Unit No. 3, DNC is proposing to change Technical Specifications 3/4.4.9.1, "Pressure/Temperature Limits," 3/4.7.2, "Steam Generator Pressure/Temperature Limitation," 3/4.7.6, "Flood Protection," 3/4.7.10, "Snubbers," 3/4.7.11, "Sealed Source Contamination," and 3/4.7.14, "Area Temperature Monitoring." Index pages viii, x, xiv and xv for Millstone Unit No. 3 will also be changed consistent with the relocation of the identified technical specifications. The Bases of the affected technical specifications will be modified to address the proposed changes.

The proposed changes will relocate selected Millstone Unit Nos. 2 and 3 technical specifications related to the Reactor Coolant System and Plant Systems to the respective Technical Requirements Manual (TRM). Information which is relocated to the TRM will be maintained in accordance with the provisions of 10 CFR 50.59.

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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 for Millstone Unit No. 2. Attachment 4 provides the retyped pages of the Technical Specifications and associated Bases for Millstone Unit No. 2. Attachment 5 provides the marked-up version of the appropriate pages of the current Technical Specifications for Millstone Unit No. 3. Attachment 6 provides the retyped pages of the Technical Specifications and associated Bases for Millstone Unit No. 3.

Environmental Considerations

DNC has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.22. DNC has determined that the proposed changes meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that the changes are being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, or that changes a surveillance requirement, and that the amendment request meets the following specific criteria.

(i) The proposed changes involve no Significant Hazards Consideration.

As demonstrated in Attachment 2, the proposed changes do not involve a Significant Hazards Consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released off site.

The proposed changes will relocate selected Millstone Unit Nos. 2 and 3 technical specifications related to the Reactor Coolant System and Plant Systems to the respective TRM. However, the operability requirements for equipment associated with these technical specifications will remain the same. The proposed changes are consistent with the design basis of the plant. The proposed changes will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not increase the type and amounts of effluents that may be released off site.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will relocate selected Millstone Unit Nos. 2 and 3 technical specifications related to the Reactor Coolant System and Plant Systems to the respective TRM. However, the operability requirements for equipment associated with these technical specifications will remain the same. The proposed changes will not result in changes in the configuration of the facility. There will be no change in the level of controls or methodology used for processing radioactive effluents or the handling of solid radioactive waste. There will be no change to the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.

Conclusions

The proposed changes were evaluated and we have concluded that they are safe. The proposed changes do not involve an 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).

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 these amendments for Millstone Unit Nos. 2 and 3 prior to January 31, 2003, with each amendment to be implemented within 90 days of issuance.

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.

J. Alan Price
Site Vice President - Millstone

Sworn to and subscribed before me

this 14th day of February, 2002

Notary Public

My Commission expires

SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachments (6)

cc: H. J. Miller, Region I Administrator
J. T. Harrison, NRC Project Manager, Unit No. 2
NRC Senior Resident Inspector, Unit No. 2
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
NRC 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 Nos. 50-336
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Attachment 1

Millstone Nuclear Power Station, Unit Nos. 2 and 3

Technical Specifications Change Requests 2-21-01 and 3-18-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Discussion of Proposed Changes

Technical Specifications Change Requests 2-21-01 and 3-18-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Discussion of Proposed Changes

Background

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating Licenses DPR-65 and NPF-49 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit Nos. 2 and 3. DNC is proposing to relocate selected technical specifications for Millstone Unit Nos. 2 and 3. For Millstone Unit No. 2, DNC is proposing to relocate part of Technical Specification 3/4.4.9.1, "Pressure/Temperature Limits," and all of Technical specifications 3/4.7.2.1, "Steam Generator Pressure/Temperature Limitation," 3/4.7.5.1, "Flood Level," 3/4.7.7.1, "Sealed Source Contamination," and 3/4.7.8, "Snubbers," to the Technical Requirements Manual (TRM) for Millstone Unit No. 2. For Millstone Unit No. 3, DNC is proposing to relocate part of Specification 3/4.4.9.1, "Pressure/Temperature Limits," and all of Technical Specifications 3/4.7.2, "Steam Generator Pressure/Temperature Limitation," 3/4.7.6, "Flood Protection," 3/4.7.10, "Snubbers," 3/4.7.11, "Sealed Source Contamination," and 3/4.7.14, "Area Temperature Monitoring," to the TRM for Millstone Unit No. 3.

The Bases of the associated technical specifications will also be moved to the Millstone Unit No. 2 or 3 TRM, as applicable. Additional background information will be included, as necessary, to explain these changes.

The Millstone Unit Nos. 2 and 3 TRMs include information which has been relocated from Technical Specifications or material which has been judged to warrant administrative control. Modifications to the TRM, which is maintained as a controlled document, are performed pursuant to the provisions of 10 CFR 50.59. The TRM is referenced by both the Millstone Unit No. 2 and 3 Final Safety Analysis Reports (FSAR).

The proposed changes are described below:

Technical Specification 3/4.4.9

The requirements of Surveillance Requirement 4.4.9.1b and Table 4.4-5 of Millstone Unit No. 2 Technical Specifications and Surveillance Requirement 4.4.9.1.2 and Table 4.4-5 of Millstone Unit No. 3 Technical Specifications will be relocated to the respective facility's TRM where future changes will be controlled in accordance with 10 CFR 50.59. The text for the surveillances will be replaced with the word "DELETED," and the tables will be deleted and replaced with, "This page intentionally left blank." The relocation of these surveillance requirements to the respective facility's TRM is in accordance with

the Standard Technical Specifications for Westinghouse and Combustion Engineering Plants,[1, 2] and the guidance provided in Generic Letter (GL) 91-01.[3]

Millstone Unit No. 2 Technical Specifications 3/4.7.2, 3/4.7.5.1, 3/4.7.7.1, 3/4.7.8, and Millstone Unit No. 3 Technical Specifications 3/4.7.2, 3/4.7.6, 3/4.7.10, 3/4.7.11, 3/4.7.14

Millstone Unit Nos. 2 and 3 Technical Specifications will be relocated to the respective facility's TRM where future changes will be controlled in accordance with 10 CFR 50.59. The text on the corresponding pages will be deleted and replaced with, "This page intentionally left blank."

Index Pages

Index pages VIII and XIII for Millstone Unit No. 2 and index pages viii, x, xiv and xv for Millstone Unit No. 3 will be revised to reflect the relocation of the identified technical specifications to the respective facility's TRM.

Technical Specification Bases

The proposed changes to the Bases for Millstone Unit Nos. 2 and 3 Surveillance Requirements 4.4.9.1b and 4.4.9.1.1 will delete the corresponding text. The proposed changes to the Bases for Millstone Unit No. 2 Technical Specifications 3/4.7.2, 3/4.7.5.1, 3/4.7.7.1, 3/4.7.8, and Millstone Unit No. 3 Technical Specifications 3/4.7.2, 3/4.7.6, 3/4.7.10, 3/4.7.11, 3/4.7.14 will delete the text associated with each section and replace the section titles with the word, "DELETED." The deleted BASES will be relocated to the respective TRM for each unit.

Safety Summary

10 CFR 50.36c(2)(ii) contains the requirements for items that must be in Technical Specifications. This regulation provides four (4) criteria that can be used to determine the requirements that must be included in the Technical Specifications. Items not meeting any of the four 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 should result in significant reductions in time and expense to modify requirements that have been

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(1) "Standard Technical Specification, Westinghouse Plants," NUREG-1431, Rev. 2, April, 2001.

(2) "Standard Technical Specification, Combustion Engineering Plants," NUREG-1432, Rev. 2, April, 2001.

(3) Nuclear Regulatory Commission, Generic Letter 91-01, "Removal of The Schedule For The Withdrawal of Reactor Vessel Material Specimens From Technical Specifications," January 4, 1991.

relocated while not adversely affecting plant safety. The criteria, and an evaluation of each technical specification proposed for relocation, are provided below.

Technical Specification 3/4.4.9

The relocation of the requirements of Surveillance Requirement 4.4.9.1b and Table 4.4-5 of Millstone Unit No. 2 and Surveillance Requirement 4.4.9.1.2 and Table 4.4-5 of Millstone Unit No. 3 to the respective facility's TRM, where future changes will be controlled in accordance with 10 CFR 50.59, is in accordance with the Standard Technical Specifications for Westinghouse and Combustion Engineering Plants, and the guidance provided in GL 91-01.

Technical Specifications (TSs) include limiting conditions for operation that establish pressure and temperature limits for the reactor coolant system. The limits are defined by TS figures that provide an acceptable range of operating temperatures and pressures for heatup, cooldown, criticality, and inservice leak and hydrostatic testing. These limits are generally valid for a specified number of effective full-power years. Reactor vessel material surveillance, through periodic withdrawal and testing of the reactor vessel material specimens, ensures the availability of data to update the inservice operating temperature and pressure limits. This surveillance duplicates the requirements of 10 CFR 50, Appendix H to monitor changes in the fracture toughness of the ferritic materials in the reactor vessel beltline.

Surveillance Requirement 4.4.9.1b of Millstone Unit No. 2 and Surveillance Requirement 4.4.9.1.2 of Millstone Unit No. 3 require the reactor vessel material irradiation surveillance specimen to be removed and examined in accordance with 10 CFR 50, Appendix H, and the results be considered in the development of the TS Figures. Improved TS 3.4.3 deletes this requirement on the basis that 10 CFR 50, Appendix H requires that the reactor vessel material irradiation surveillance specimen to be removed and examined to monitor changes in fracture toughness properties and that the information be used to update pressure and temperature limits in the TS. Inclusion of these requirements in the TS constitutes a duplication of the requirements outlined in 10 CFR 50, Appendix H. Millstone Unit Nos. 2 and 3 are required to comply with the provisions of 10 CFR 50, Appendix H, therefore, these Surveillance Requirements are not required to be in the TS. The relocation of these surveillance requirements to the respective facility's TRM is consistent with the Standard TSs for Westinghouse and Combustion Engineering Plants (NUREG-1431 and NUREG-1432), and the guidance provided in GL 91-01.

Technical Specification 3/4.7.2.1 for Millstone Unit No. 2 and 3/4.7.2 for Millstone Unit No. 3

TS 3/4.7.2.1 for Millstone Unit No. 2 and 3/4.7.2 for Millstone Unit No. 3 are proposed to be relocated to the respective facility's TRM. These specifications provide limitation on steam generator pressure and temperature which ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture

toughness stress limits. For Millstone Unit No. 2, the limitations of 70 °F and 200 psig are based on a steam generator nil ductility transition temperature (RT_{NDT}) of 50 °F and are sufficient to prevent brittle fracture. For Millstone Unit No. 3, the limitations of 70 °F and 200 psig are based on a steam generator RT_{NDT} of 60 °F and are sufficient to prevent brittle fracture.

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.

The limitations imposed by these two specifications on steam generators temperature and pressure are provided to prevent brittle fracture in the steam generators and do not describe a limitation on instrumentation that is used to detect, and indicate in the control room, an abnormal degradation of the reactor coolant pressure boundary. These specifications do not meet Criterion 1.

Criterion 2

A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The limitations imposed by these two specifications on steam generators temperature and pressure are provided to prevent brittle fracture in the steam generators. The limiting temperature and pressure values described in these specifications do not provide direct input to Reactor Protection System or Engineered Safety Features Actuation System functions, nor do they represent process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3

A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

These specifications do cover a component (steam generator) that is part of the primary success path (heat removal) 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. However, the specification limits apply to conditions when the Reactor Coolant System temperature is

unusually low (< 70°F). Under these conditions, the Steam Generators are not required to function to mitigate any DBA or transient. Therefore, this specification does not satisfy criterion 3.

The Steam Generator heat removal function is covered in the Reactor Coolant Loop specifications.

<u>Criterion 4</u>

A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The limits covered by these specifications have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. This TS does not cover a SSC requiring risk review/unavailability monitoring. This specification does not meet Criterion 4.

These TSs do not fulfill any one or more of the 10 CFR 50.36c(2)(ii) criteria on items for which TSs must be established. Therefore, these TSs can be relocated to the TRM.

<u>Technical Specification 3/4.7.5.1 for Millstone Unit No. 2 and 3/4.7.6 for Millstone Unit No. 3</u>

TS 3/4.7.5.1 for Millstone Unit No. 2 and 3/4.7.6 for Millstone Unit No. 3 are proposed to be relocated to the respective facility's TRM. TS 3/4.7.5.1 for Millstone Unit No. 2 ensures that one service water pump motor will be protected against flooding to a minimum elevation of 28 feet. The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. Action to provide pump motor protection will be initiated when the water level reaches plant grade level in order to ensure operator accessibility to the intake structure. TS 3/4.7.6 for Millstone Unit No. 3 ensures that the service water pump cubicle watertight doors will be closed and the pump cubicle sump drain valves will be closed before the water level reaches the critical elevation of 14.5 feet Mean Sea Level (MSL). Elevation 14.5 feet MSL is the floor elevation of the service water pump cubicle.

<u>Criterion 1</u>

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

These TSs cover protective actions against flooding of the service water pump motors. They do not cover installed instrumentation which is used to detect, and

indicate in the control room, an abnormal degradation of the reactor coolant pressure boundary. These specifications do not meet Criterion 1.

Criterion 2

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

These TSs cover protective actions against flooding of the service water pump motors. They do not cover 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. These specifications do not satisfy criterion 2.

Criterion 3

A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

These TSs ensure that service water pump motors are protected against flooding conditions. These TSs do not affect the ability of a SSC, that is part of the primary success path and which functions or actuates to mitigate a design basis accident or a transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier, from performing its safety function. These specifications do not satisfy criterion 3.

Criterion 4

A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The limitations covered by these TSs have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. These specifications do not meet Criterion 4.

These TSs do not fulfill any one or more of the 10 CFR 50.36c(2)(ii) criteria on items for which TSs must be established. Therefore, these TSs can be relocated to the TRM.

Technical Specification 3/4.7.8 for Millstone Unit No. 2 and 3/4.7.10 for Millstone Unit No. 3

TS 3/4.7.8 for Millstone Unit No. 2 and 3/4.7.10 for Millstone Unit No. 3 are proposed to be relocated to the respective facility's TRM. The snubbers are required to be operable to ensure the structural integrity of the Reactor Coolant System and all other safety

related systems is maintained during and following a seismic or other event initiating dynamic loads. The restraining action of the snubbers ensures that the initiating event failure does not propagate to other parts of the failed system or to other safety systems. Snubbers also allow thermal expansion of piping and nozzles to eliminate excessive thermal stresses during heatup or cooldown.

<u>Criterion 1</u>

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

These TSs 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 specifications do not satisfy criterion 1.

<u>Criterion 2</u>

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

These TSs only require snubbers to be operable. Snubbers are not a design feature 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. These specifications do not satisfy criterion 2.

<u>Criterion 3</u>

A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

These TSs only require snubbers to be operable. Snubbers are not required to mitigate any DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. These specifications do not satisfy criterion 3.

<u>Criterion 4</u>

A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Snubbers have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. These specifications do not meet Criterion 4.

These TSs do not fulfill any one or more of the 10 CFR 50.36c(2)(ii) criteria on items for which TSs must be established. Therefore, these TSs can be relocated to the TRM.

Technical Specification 3/4.7.7 for Millstone Unit No. 2 and 3/4.7.11 for Millstone Unit No. 3

TS 3/4.7.7 for Millstone Unit No. 2 and 3/4.7.11 for Millstone Unit No. 3 are proposed to be relocated to the respective facility's TRM. The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested. Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are not frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

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.

These TSs provide limitations on sealed source removable contamination to ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. Therefore, these TSs 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 specifications do not satisfy criterion 1.

Criterion 2

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

These TSs ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. They are not applicable to a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, these specifications do not meet Criterion 2.

Criterion 3

A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

These TSs cover limitations on sealed source removable contamination to ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. They do not cover a structure, system, or component that is part of the primary success path 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. Therefore, these specifications do not satisfy criterion 3.

Criterion 4

A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The limitations covered by these TSs have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. These specifications do not meet Criterion 4.

These TSs do not fulfill any one or more of the 10 CFR 50.36c(2)(ii) criteria on items for which technical specifications must be established. Therefore, these TSs can be relocated to the TRM.

Technical Specification 3/4.7.14 for Millstone Unit No. 3

TS 3/4.7.14 for Millstone Unit No. 3 is proposed to be relocated to the TRM. The area temperature limitations imposed by this TS ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±2.2°F.

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 TS 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 specifications do not satisfy criterion 1.

Criterion 2

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

Area Temperatures are not process variables that are an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. These temperature limits are requirements of the Equipment Environmental Qualification (EEQ) program. Area Temperatures do not satisfy criterion 2.

Criterion 3

A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Area Temperature Monitoring is not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Area Temperature monitoring does not satisfy criterion 3.

Criterion 4

A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The limitations covered by this TS have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. This specification does not meet Criterion 4.

This TS does not fulfill any one or more of the 10 CFR 50.36c(2)(ii) criteria on items for which TSs must be established. Therefore, this TS can be relocated to the TRM.

Index Pages

Revision of Index Pages VIII and XIII for Millstone Unit No. 2 and Index Pages viii, x, xiv and xv for Millstone Unit No. 3 are administrative changes. These changes are consistent with the changes previously discussed. Therefore, the proposed changes will have no adverse effect on plant safety.

Technical Specification Changes - Bases

The information contained in the Bases of the affected TSs will not be modified as a result of the proposed TS changes. The proposed changes will not result in any new approaches to plant operation. Therefore, the proposed Bases changes will not adversely affect public safety.

The relocation of the requirements for the applicable TSs to the TRM will not result in any new approaches to plant operation and will not adversely affect any accident mitigation equipment. The plant response to the DBAs will not change. Therefore, the proposed changes will not adversely affect public health and safety. Thus, the proposed changes are safe.

Docket Nos. 50-336
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Attachment 2

Millstone Nuclear Power Station, Unit Nos. 2 and 3

Technical Specifications Change Requests 2-21-01 and 3-18-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Significant Hazards Consideration

Technical Specifications Change Requests 2-21-01 and 3-18-01
Relocation of Selected Technical Specifications Related
to the Reactor Coolant System and Plant Systems
Significant Hazards Consideration

Description of License Amendment Request

Dominion Nuclear Connecticut, Inc. (DNC) is proposing to relocate selected technical specifications for Millstone Unit Nos. 2 and 3. For Millstone Unit No. 2, DNC is proposing to relocate part of Technical Specification 3/4.4.9.1, "Pressure/Temperature Limits," and all of Technical Specifications 3/4.7.2.1, "Steam Generator Pressure/Temperature Limitation," 3/4.7.5.1, "Flood Level," 3/4.7.7.1, "Sealed Source Contamination," and 3/4.7.8, "Snubbers," to the Technical Requirements Manual (TRM) for Millstone Unit No. 2. For Millstone Unit No. 3, DNC is proposing to relocate part of Specification 3/4.4.9.1, "Pressure/Temperature Limits," and all of Technical Specifications 3/4.7.2, "Steam Generator Pressure/Temperature Limitation," 3/4.7.6, "Flood Protection," 3/4.7.10, "Snubbers," 3/4.7.11, "Sealed Source Contamination," and 3/4.7.14, "Area Temperature Monitoring," to the TRM for Millstone Unit No. 3.

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 would not:

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

   The proposed technical specification changes will relocate to the TRM the following items: surveillance requirements for the withdrawal of reactor vessel material irradiation specimens of Millstone Unit Nos. 2 and 3 which are part of the Pressure/Temperature Limits technical specifications, Millstone Unit Nos. 2 and 3 technical specifications covering Steam Generator Pressure/Temperature Limitation, Flood Level, Sealed Source Contamination, and Snubbers. Also the Millstone Unit No. 3 technical specification covering Area Temperature Monitoring will be relocated to the TRM. Since the relocated requirements remain the same, the proposed changes will have no effect on plant operation, or the availability or operation of any accident mitigation equipment. Therefore, the relocation of the requirements associated with these technical specifications will not impact an accident initiator and cannot cause an accident. These 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 technical specification changes will relocate the requirements of selected Millstone Unit Nos. 2 and 3 technical specifications as described above to the TRM. The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. Since the requirements remain the same, the proposed changes do not alter the way any system, structure, or component functions and do not alter the manner in which the plant is operated. 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 proposed technical specification changes will relocate to the TRM the following items: surveillance requirements for the withdrawal of reactor vessel material irradiation specimens of Millstone Unit Nos. 2 and 3 which are part of the Pressure/Temperature Limits technical specifications, Millstone Unit Nos. 2 and 3 technical specifications covering Steam Generator Pressure/Temperature Limitation, Flood Level, Sealed Source Contamination, and Snubbers. Also the Millstone Unit No. 3 technical specification covering Area Temperature Monitoring will be relocated to the TRM. Since the proposed changes are solely to relocate the existing requirements, the proposed changes will have no effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the Design Basis Accidents will not change. Therefore, there will be no reduction in a margin of safety.

Docket Nos. 50-336
50-423
B18556

Attachment 3

Millstone Nuclear Power Station, Unit No. 2

Technical Specifications Change Request 2-21-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Marked Up Pages

List of Affected Pages

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
Index Page VIII		Amend. No. 238
Index Page XIII		Amend. No. 240
3/4.4.9.1	Pressure/Temperature Limits	3/4 4-18, Amend. No. 218 3/4 4-20, Amend. No. 94
3/4.7.2.1	Steam Generator Pressure/Temperature Limitation	3/4 7-10, Original issue
3/4.7.5.1	Flood Level	3/4 7-13, Amend. No. 101 3/4 7-14, Original issue 3/4 7-15, Original issue
3/4.7.7.1	Sealed Source Contamination	3/4 7-19, Amend. No. 202 3/4 7-20, Original issue
3/4.7.8	Snubbers	3/4 7-21, Amend. No. 160 3/4 7-22, Amend. No. 244 3/4 7-22a, Amend. No. 239 3/4 7-22b, Amend. No. 160 3/4 7-32, Amend. No. 160
3/4.4.9 Bases	Pressure/Temperature Limits	B 3/4 4-7, Amend. No. 218
3/4.7.2 Bases	Steam Generator Pressure/Temperature Limitation	B 3/4 7-3a, Amend. No. 238
3/4.7.5 Bases	Flood Level	B 3/4 7-4a, Amend. No. 236
3/4.7.7 and 3/4.7.8 Bases	Sealed Source Contamination and Snubbers	B 3/4 7-5, Amend. No. 202, B 3/4 7-6, Amend. No. 244

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

For Information only
No Change

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 Reactor Coolant System (except the pressurizer) temperature, pressure, and heatup and cooldown rates shall be limited in accordance with the limits specified in Table 3.4-2 and shown on Figures 3.4-2a and 3.4-2b.

APPLICABILITY: At all times.*

ACTION:

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 300 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.

*See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.9.1

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

b. ~~The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-3. The results of these examinations shall be used to update Table 3.4-2 and Figures 3.4-2a and 3.4-2b.~~

DELETED

For Information only

No Change

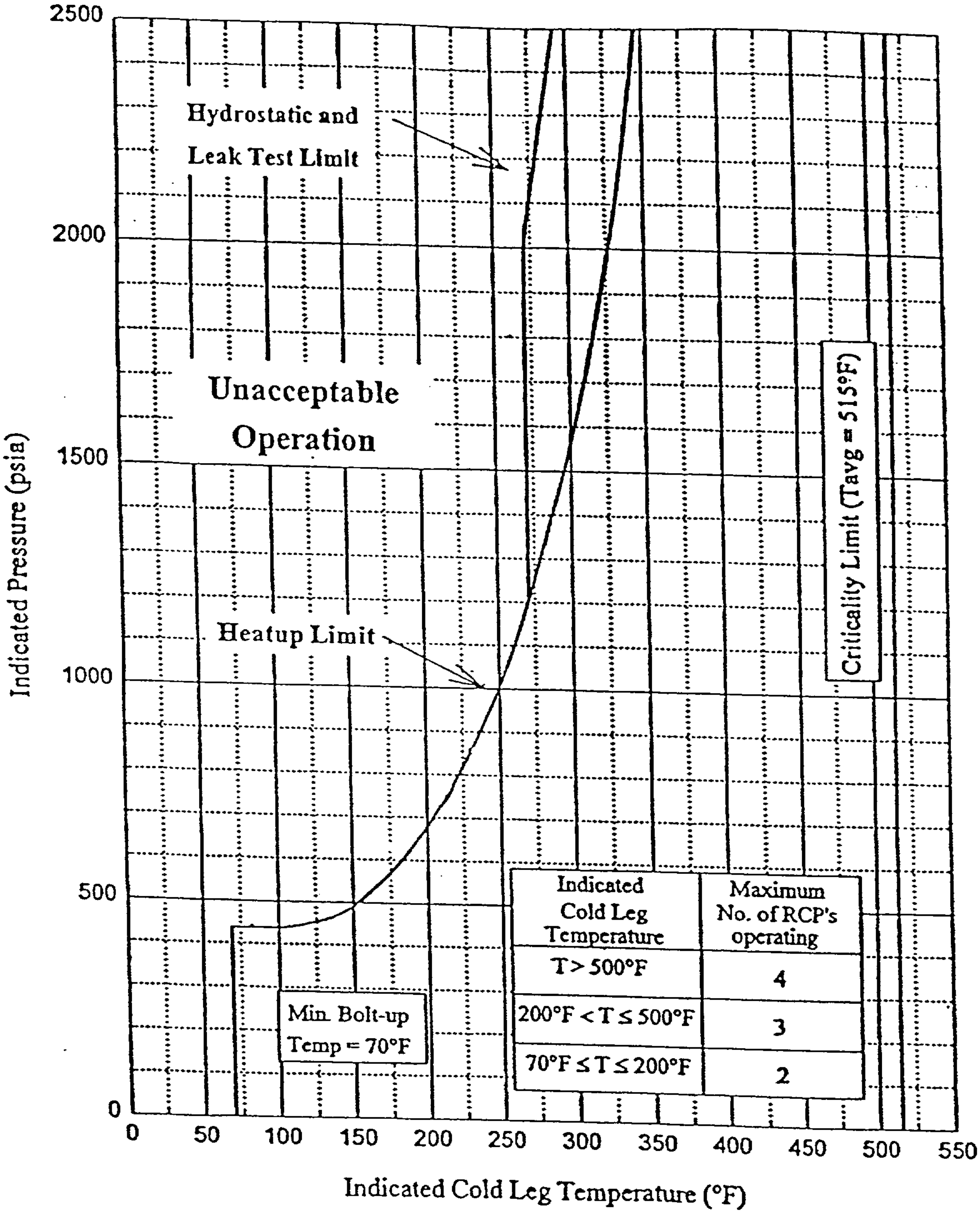
July 1, 1998

TABLE 3.4-2

REACTOR COOLANT SYSTEM
HEATUP AND COOLDOWN LIMITS

Cooldown		Heatup	
Indicated Cold Leg Temperature	Limit	Indicated Cold Leg Temperature	Limit
$\leq$ 100°F	$\leq$ 5°F/hour if RCS not vented.	$\leq$ 220°F	$\leq$ 30°F/hour
100°F < T $\leq$ 230°F	$\leq$ 30°F/hour if RCS not vented.	220°F < T $\leq$ 275°F	$\leq$ 50°F/hour
< 190°F	$\leq$ 50°F/hour if RCS vent $\geq$ 2.2 square inches.	> 275°F	$\leq$ 100°F/hour
$\leq$ 230°F	$\leq$ 50°F/hour during unanticipated temperature excursions.		
> 230°F	$\leq$ 80°F/hour		
		Inservice Hydrostatic and Leak Testing	
		Indicated Cold Leg Temperature	$\leq$ 5°F/hour for 1 hour prior to and during inservice hydrostatic and leak testing operations above the heatup limit curve.

For Information Only
No change

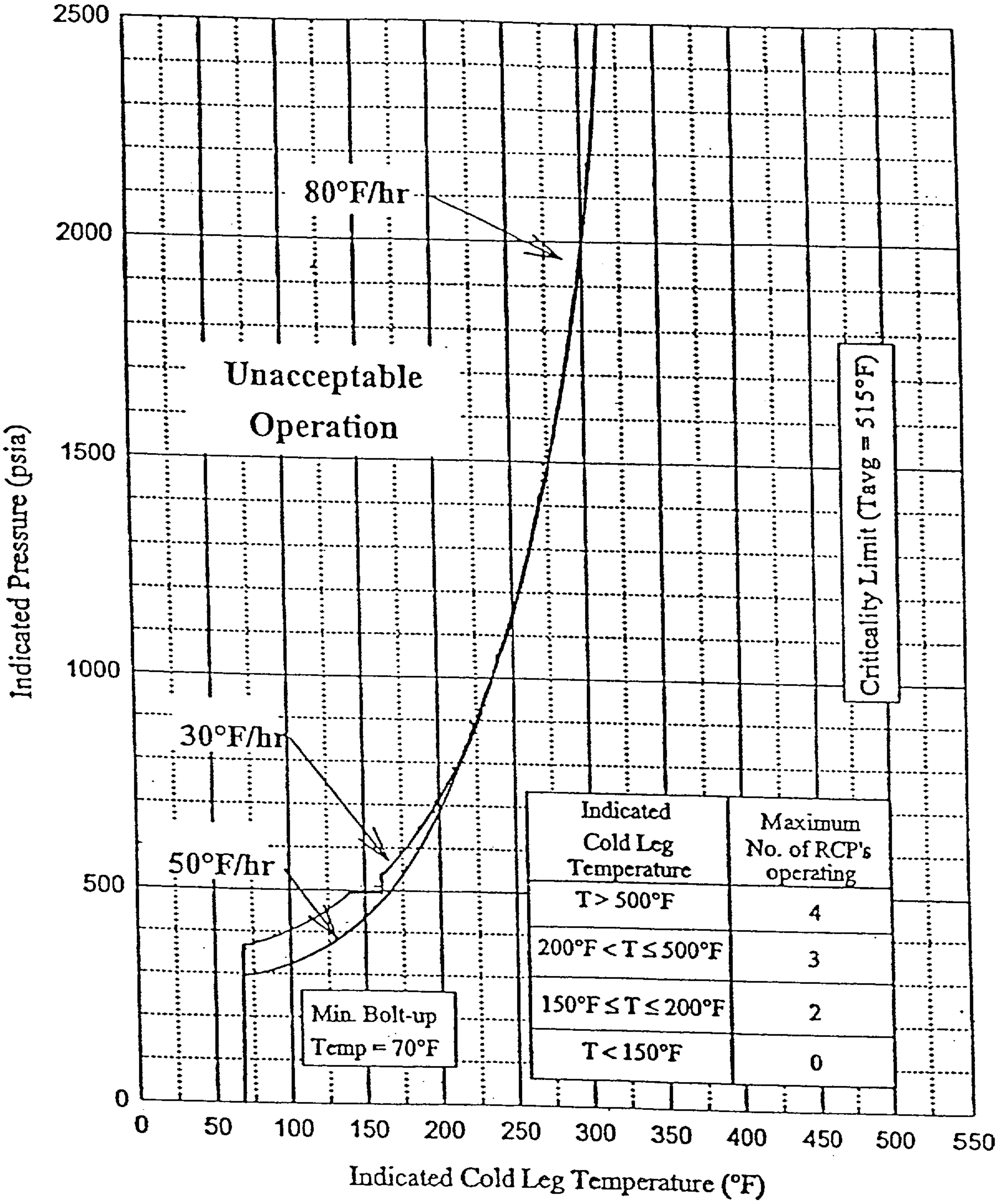


Indicated Cold Leg Temperature	Maximum No. of RCP's operating
T > 500°F	4
200°F < T ≤ 500°F	3
70°F ≤ T ≤ 200°F	2

Millstone Unit 2 Reactor Coolant System
Heatup Limitations for Up to 20 EFPY
Figure 3.4-2a

For Information Only
No Change

July 1, 1998



Indicated Cold Leg Temperature	Maximum No. of RCP's operating
T > 500°F	4
200°F < T ≤ 500°F	3
150°F ≤ T ≤ 200°F	2
T < 150°F	0

Millstone Unit 2 Reactor Coolant System
Cooldown Limitations for Up to 20 EFPY
Figure 3.4-2b

~~TABLE 4.4-3~~

~~REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE~~

CAPSULE	SCHEDULE (EFPY)
W-97	3.0
W-104	10.0
W-284	17.0
W-263	24.0
W-277	32.0
W-83	Spare
W-97 (Flux Monitor)	10.0

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PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied:

a. Immediately reduce the steam generator pressure to ≤ 200 psig, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be determined to be > 70°F at least once per hour when pressures in the steam generators are > 200 psig and T_{avg} is < 200°F.

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3/4.7.5 FLOOD LEVEL

LIMITING CONDITION FOR OPERATION

3.7.5.1 At least one OPERABLE service water pump motor shall be protected against flooding to a minimum elevation of 28 feet Mean Sea Level USGS datum if either:

a. The water level, including wave crest height, is exceeding plant grade level (14.0 feet Mean Sea Level USGS datum), or

b. Three or more of the following conditions are occurring simultaneously:

1. The center of a storm, as determined by radar, reconnaissance or forecasted track projection, is presently located within the critical area as defined on Figure 3.7-1.

2. The projected track of a storm approaching the facility as determined by radar, reconnaissance or forecasted track projection, lies between 130° and 350°.

3. The central pressure of the storm is or is forecasted to be ≤ 28.0 in. Hg; or the measured 15 minute average wind speed at nominal elevation 389 on the meteorological tower exceeds 60 mph.

4. The 15 minute average wind direction at nominal elevation 389 on the meteorological tower is within the sector from 150° clockwise to 300°.

APPLICABILITY: ALL MODES

ACTION:

With the water level exceeding either plant grade or with three or more of the above specified meteorological conditions being exceeded simultaneously, immediate initiate action to protect at least one service water pump motor against flooding to a minimum elevation of 28 feet; complete this protective action within 2 hours.

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.5.1.1 The water level shall be determined to be below plant grade at least once per hour when the eye of a hurricane is within 150 miles of the facility.

4.7.5.1.2 The above specified meteorological conditions shall be determined at least once per 2 hours when a hurricane eye is within 150 miles of the facility. The meteorological conditions shall be determined from forecasts obtained from the Connecticut Valley Electrical Exchange (CONVEX) and from the site meterological instrumentation.

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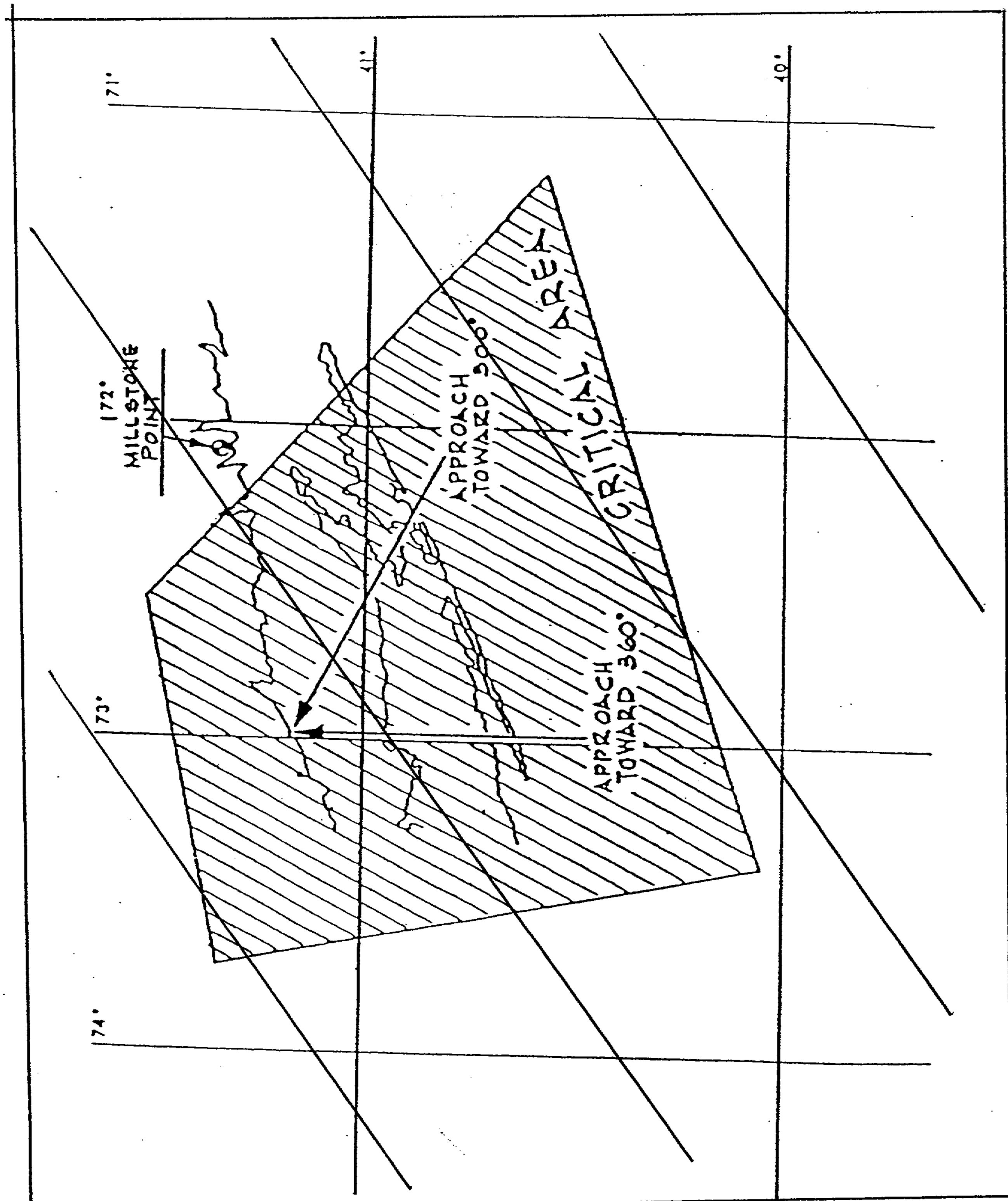
~~August 1, 1975~~



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Critical Area
Figure 3.7-1

add

PLANT SYSTEMS

3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.7.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: AT ALL TIMES.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.7.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequencies described below.

a. Sources in use (excluding startup sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive material:

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~~August 1, 1975~~

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources - Each sealed startup source shall be tested prior to being subjected to core flux and following repair or maintenance to the source.

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3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8.d on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for the system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-3. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-3 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 160.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type

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Corrected by letter dated 8/26/92

that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8.e or 4.7.8.f, as applicable. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Snubber Tests

At least once per eighteen (18) months during shutdown, a representative sample (10% of the total of each type of snubber, mechanical and hydraulic, in use in the plant) shall be tested either in place or in a bench test. For each snubber that does not meet the test acceptance criteria of Specification 4.7.8.e or 4.7.8.f, as applicable, an additional 5% of that type of snubber shall be tested.

Testing shall continue until no additional inoperable snubbers are found within a sample or until all snubbers have been tested. The representative sample selected for testing shall include the various configurations, and the range of size and capacity of snubbers.

Snubbers identified as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

In addition to the regular sample, in locations where snubbers had failed the previous test due to operational or environmental conditions (excessive vibration, water hammer, high radiation, extreme heat or humidity, etc.), the snubbers currently installed in these locations shall be tested during the next test period. Test results of these snubbers may not be included for the resampling. All replacement snubbers shall have been tested prior to installation.

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*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented.

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SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for testing either fails to lock-up or fails to move (i.e., frozen in place), the cause will be evaluated and if caused by manufacturer design deficiency, all snubbers of the same design subject to the same defect shall be tested regardless of location or difficulty or removal. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber (s) were adversely affected by the inoperability of the snubber (s) in order to ensure that the supported component remains capable of meeting the designed service.

e. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

f. Mechanical Snubbers Functional Test Acceptance Criteria*

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

g. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Quality Assurance Program Topical Report.

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*Mechanical snubber functional test acceptance criteria shall become effective upon installation of snubber testing equipment but not later than June 30, 1985.

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENT (Continued)

~~Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.~~

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TABLE 4.7-3
SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and that decision shall be used as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

Reducing T_{avg} to < 515°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary 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 primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with iodine spiking phenomena. 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

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.0 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. In addition, during heatup and cooldown evolutions, the RCS ferritic materials transition between ductile and brittle (non-ductile) behavior. To provide adequate protection, the pressure/temperature limits were developed in accordance with the 10CFR50 Appendix G requirements to ensure the margins of safety against non-ductile failure are maintained during all normal and anticipated operational occurrences. These pressure/temperature limits are provided in Figures 3.4-2a and 3.4-2b and the heatup and cooldown rates are contained in Table 3.4-2.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermally induced compressive stresses at the inside wall tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure- temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermally induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

For Information Only

No change

The heatup and cooldown limit curves (Figures 3.4-2a and 3.4-2b) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to the maximums described in Technical Specification 3.4.9.1, Table 3.4-2. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2a and 3.4-2b.

Verification that RCS pressure and temperature conditions are within the limits of Figures 3.4-2a and 3.4-2b and Table 3.4-2, at least once per 30 minutes, is required when undergoing planned changes of $\geq$ 10°F or $\geq$ 100 psi. This frequency is considered reasonable since the location of interest during cooldown is over two inches (i.e. 1/4 t location) from the interface with the reactor coolant. During heatup the location of interest is over six inches from the interface with the reactor coolant. This combined with the relatively large heat retention capability of the reactor vessel ensures that small temperature fluctuations such as those expected during normal heatup and cooldown evolutions do not challenge the structural integrity of the reactor vessel when monitored on a 30 minute frequency. The 30 minute time interval permits assessment and correction for minor deviations within a reasonable time.

During RCS heatup and cooldown the magnitude of the stresses across the reactor vessel wall are controlled by restricting the rate of temperature change and the system pressure. The RCS pressure/temperature limits are provided in Figures 3.4-2a and 3.4-2b, and the heatup and cooldown rates are contained in Table 3.4-2. The following guidelines should be used to ensure compliance with the Technical Specification limits.

1. When changing RCS temperature, with any reactor coolant pumps in operation, the rate of temperature change is calculated by using RCS loop cold leg temperature indications.

   This also applies during parallel reactor coolant pump and shutdown cooling (SDC) pump operation because the RCS loop cold leg temperature is the best indication of the temperature of the fluid in contact with the reactor vessel wall. Even though SDC return temperature may be below RCS cold leg temperature, the mixing of a large quantity of RCS cold leg water and a small quantity of SDC return water will result in the temperature of the water reaching the reactor vessel wall being very close to RCS cold leg temperature.

2. When changing RCS temperature via natural circulation, the rate of temperature change is calculated by using RCS loop cold leg temperature indications.

3. When changing RCS temperature with only SDC in service, the rate of temperature change is calculated by using SDC return temperature indication.

4. During the transition from natural circulation flow, to forced flow with SDC pumps, the rate of temperature change is calculated by using RCS loop cold leg temperature indications. SDC return temperature should be used to calculate the rate of temperature change after SDC is in service, RBCCW flow has been established to the SDC heat exchanger(s), and SDC return temperature has decreased below RCS cold leg temperature.

5. During the transition from parallel reactor coolant pump and SDC pump operation, the rate of temperature change is calculated by using RCS loop cold leg temperature indications until all reactor coolant pumps are secured. SDC return temperature should be used to calculate the rate of temperature change after all reactor coolant pumps have been secured.

6. The temperature change limits are for a continuous one hour period. Verification of operation within the limit must compare the current RCS water temperature to the value that existed one hour before the current time. If the maximum temperature increase or decrease, during this one hour period, exceeds the Technical Specification limit, appropriate action should be taken.

7. When a new, more restrictive temperature change limit is approached, it will be necessary to adjust the current temperature change rate such that as soon as the new rate applies, the total temperature change for the previous one hour does not exceed the new more restrictive rate.

   The same principle applies when moving from one temperature change limit curve to another. If the new curve is above the current curve (higher RCS pressure for a given RCS temperature), the new curve will reduce the temperature change limit. It will be necessary to first ensure the new more restrictive temperature change limit will not be exceeded by looking at the total RCS temperature change for the previous one hour time period. If the magnitude of the previous one hour temperature change will exceed the new limit, RCS temperature should be stabilized to allow the thermal stresses to dissipate. This may require up to a one hour soak before RCS pressure may be raised within the limits of the new curve.

   If the new curve is below the current curve (lower RCS pressure for a given RCS temperature), the new curve will allow a higher temperature change limit. All that is necessary is to lower RCS pressure, and then apply the new higher temperature change limit.

8. When performing evolutions that may result in rapid and significant temperature swings (e.g. placing SDC in service or shifting SDC heat exchangers), the total temperature change limit for the previous one hour period must not be exceeded. If a significant temperature change is anticipated, and an RCS heatup or cooldown is in progress, the plant should be stabilized for up to one hour, before performing this type of evolution. Stabilizing the plant for up to one hour will allow the thermal stresses, from any previous RCS temperature change, to dissipate. This will allow rapid RCS temperature changes up to the applicable Technical Specification temperature change limit.

For Information Only

No Change

9. Additional margin, to prevent exceeding the Appendix G limits when RCS temperature is at or below 230°F, can be obtained by maintaining RCS pressure below the pressure allowed by the 50°F/hr cooldown curve provided on Figure 3.4-2b. This will ensure that if a greater than anticipated temperature excursion occurs during short duration evolutions, the margins of safety required by Appendix G will not be exceeded. Examples of plant evolutions that may result in unanticipated temperature excursions include placing SDC in service without parallel RCP operation, securing RCPs when SDC is already in service, shifting SDC heat exchangers, and switching SDC pumps. Establishing a lower RCS pressure, will minimize the probability of exceeding Appendix G limits.

   If the 50°F/hr cooldown curve is used to evaluate unanticipated temperature excursions while limited to a cooldown rate of 30°F/hr, the RCS cooldown rate must be restored to within the 30°F/hr limit as soon as practical. This may require a soak period to allow the thermal stresses, from the previous RCS temperature change, to dissipate.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table 4.6-1 of the Final Safety Analysis Report. Reactor operation and resultant fast neutron irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, can be predicted using the methods described in Revision 2 to Regulatory Guide 1.99.

The heatup and cooldown limit curves shown on Figures 3.4-2a and 3.4-2b include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible uncertainties in the pressure and temperature sensing instruments. The adjustments include the pressure and temperature instrument and loop uncertainties associated with the main control board displays, the pressure drop across the core (RCP operation), and the elevation differences between the location of the pressure transmitters and the vessel beltline region. In addition to these curve adjustments, the LTOP evaluation includes adjustments due to valve stroke times, PORV circuitry reaction times, and valve discharge backpressure.

The actual shift in RT_{NDT} of the vessel material is established periodically during operation by removing and evaluating, in accordance with 10CFR50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are similar, the measured transition shift for a sample can be correlated to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. Operation of the RCS within the limits of the heatup and cooldown curves will ensure compliance with this requirement.

Included in this evaluation is consideration of flange protection in accordance with 10 CFR 50, Appendix G. The requirement makes the minimum temperature RT_{NDT} plus 90°F for hydrostatic test and RT_{NDT} plus 120°F for normal operation when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure. Since the flange region RT_{NDT} has been calculated to be 30°F, the minimum flange pressurization temperature during normal operation is 150°F (161°F with instrument uncertainty) when the pressure exceeds 20% of the preservice hydrostatic pressure. Operation of the RCS within the limits of the heatup and cooldown curves will ensure compliance with this requirement.

To establish the minimum boltup temperature, ASME Code Section XI, Appendix G, requires the temperature of the flange and adjacent shell and head regions shall be above the limiting RT_{NDT} temperature for the most limiting material of these regions. The RT_{NDT} temperature for that material is 30°F. Adding 10.5°F, for temperature measurement uncertainty, results in a minimum boltup temperature of 40.5°F. For additional conservatism, a minimum boltup temperature of 70°F is specified on the heatup and cooldown curves. The head and vessel flange region temperature must be greater than 70°F, whenever any reactor vessel stud is tensioned.

~~The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50. Removal of reactor vessel irradiation surveillance specimens does not constitute a CORE ALTERATION per Specification 1.12.~~

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements. Verification that pressurizer temperature conditions are within the limits of LCO 3.4.9.2, at least once per 30 minutes, is required when undergoing planned changes of ≥ 10°F. The 30 minute time interval permits assessment and correction for temperature deviations within a reasonable time.

The Low Temperature Overpressure Protection (LTOP) System provides a physical barrier against exceeding the 10CFR50 Appendix G pressure/temperature limits during low temperature RCS operation either with a steam bubble in the pressurizer or during water solid conditions. This system consists of either two PORVs (each PORV is equivalent to a vent of approximately 1.4 square inches) with a pressure setpoint $\leq$ 415 psia, or an RCS vent of sufficient size. Analysis has confirmed that the design basis mass addition transient discussed below will be mitigated by operation of the PORVs or by establishing an RCS vent of sufficient size.

The LTOP System is required to be OPERABLE when RCS cold leg temperature is at or below 275°F (Technical Specification 3.4.9.3). However, if the RCS is in MODE 6 and the reactor vessel head has been removed, a vent of sufficient size has been established such that RCS pressurization is not possible. Therefore, an LTOP System is not required (Technical Specification 3.4.9.3 is not applicable).

The LTOP System is armed at a temperature which exceeds the limiting 1/4t RT_{NDT} plus 90°F as required by NUREG-0800 (i.e., SRP), Branch Technical Position RSB 5-2. For the operating period up to 20 EFPY, the limiting 1/4t RT_{NDT} is 145°F which results in a minimum LTOP System enable temperature of at least 263°F when corrected for instrument uncertainty. The current value of 275°F will be retained.

The mass input analysis performed to ensure the LTOP System is capable of protecting the reactor vessel assumes that all pumps capable of injecting into the RCS start, and then one PORV fails to actuate (single active failure). Since the PORVs have limited relief capability, certain administrative restrictions have been implemented to ensure that the mass input transient will not exceed the relief capacity of a PORV. The analysis has determined two PORVs (assuming one PORV fails) are sufficient if the mass addition transient is limited to the inadvertent start of one high pressure safety injection (HPSI) pump and two charging pumps when RCS temperature is at or below 275°F and above 190°F, and the inadvertent start of one charging pump when RCS temperature is at or below 190°F.

The assumed active failure of one PORV results in an equivalent RCS vent size of approximately 1.4 square inches when the one remaining PORV opens. Therefore, a passive vent of at least 1.4 square inches can be substituted for the PORVs. However, a vent size of at least 2.2 square inches will be required when venting the RCS. If the RCS is depressurized and vented through at least a 2.2 square inch vent, the peak RCS pressure, resulting from the maximum mass input transient allowed by Technical Specification 3.4.9.3, will not exceed 300 psig (SDC System suction side design pressure).

When the RCS is at or below 190°F, additional pumping capacity can be made capable of injecting into the RCS by establishing an RCS vent of at least 2.2 square inches. Removing a pressurizer PORV or the pressurizer manway will result in a passive vent of at least 2.2 square inches. Additional methods to establish the required RCS vent are acceptable, provided the proposed vent has been evaluated to ensure the flow characteristics are equivalent to one of these.

Establishing a pressurizer steam bubble of sufficient size will be sufficient to protect the reactor vessel from the energy addition transient associated with the start of an RCP, provided the restrictions contained in Technical Specification 3.4.1.3 are met. These restrictions limit the heat

For Information only

No change

input from the secondary system. They also ensure sufficient steam volume exists in the pressurizer to accommodate the insurge. No credit for PORV actuation was assumed in the LTOP analysis of the energy addition transient.

The restrictions apply only to the start of the first RCP. Once at least one RCP is running, equilibrium is achieved between the primary and secondary temperatures, eliminating any significant energy addition associated with the start of the second RCP.

The LTOP restrictions are based on RCS cold leg temperature. This temperature will be determined by using RCS cold leg temperature indication when RCPs are running, or natural circulation if it is occurring. Otherwise, SDC return temperature indication will be used.

Restrictions on RCS makeup pumping capacity are included in Technical Specification 3.4.9.3. These restrictions are based on balancing the requirements for LTOP and shutdown risk. For shutdown risk reduction, it is desirable to have maximum makeup capacity and to maintain the RCS full (not vented). However, for LTOP it is desirable to minimize makeup capacity and vent the RCS. To satisfy these competing requirements, makeup pumps can be made not capable of injecting, but available at short notice.

A charging pump can be considered to be not capable of injecting into the RCS by use of any of the following methods and the appropriate administrative controls.

1. Placing the motor circuit breaker in the open position.
2. Removing the charging pump motor overload heaters from the charging pump circuit.
3. Removing the charging pump motor controller from the motor control center.

A HPSI pump can be considered to be not capable of injecting into the RCS by use of any of the following methods and the appropriate administrative controls.

1. Racking down the motor circuit breaker from the power supply circuit.
2. Shutting and tagging the discharge valve with the key lock on the control panel (2-SI-654 or 2-SI-656).
3. Placing the pump control switch in the pull-to-lock position and removing the breaker control power fuses.
4. Placing the pump control switch in the pull-to-lock position and shutting the discharge valve with the key lock on the control panel (2-SI-654 or 2-SI-656).

These methods to prevent charging pumps and HPSI pumps from injecting into the RCS, when combined with the appropriate administrative controls, meet the requirement for two independent means to prevent pump injection as a result of a single failure or inadvertent single action.

For Information Only

These methods prevent inadvertent pump injections while allowing manual actions to rapidly restore the makeup capability if conditions require the use of additional charging or HPSI pumps for makeup in the event of a loss of RCS inventory or reduction in shutdown margin.

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 Appendix G 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.

An exception to Technical Specification 3.0.4 is specified for Technical Specification 3.4.9.3 to allow a plant cooldown to MODE 5 if one or both PORVs are inoperable. MODE 5 conditions may be necessary to repair the PORV(s).

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a.

a feedwater isolation signal since the steam line break accident analysis credits them in prevention of feed line volume flashing in some cases. Feedwater pumps are assumed to trip immediately with an MSI signal.

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

The atmospheric dump valve (ADV) lines provide a method to maintain the unit in HOT STANDBY, and to replace or supplement the condenser steam dump valves to cool the unit to Shutdown Cooling (SDC) entry conditions. Each ADV line contains an air operated ADV, and an upstream manual isolation valve. The manual isolation valves are normally open, and the ADVs closed. The ADVs, which are normally operated from the main control room, can be operated locally using a manual handwheel.

An ADV line is OPERABLE if local manual operation of the associated valves can be used to perform a controlled release of steam to the atmosphere. This is consistent with the LOCA analysis which credits local manual operation of the ADV lines for accident mitigation.

3/4.7.1.8 STEAM GENERATOR BLOWDOWN ISOLATION VALVES

The steam generator blowdown isolation valves will isolate steam generator blowdown on low steam generator water level. An auxiliary feedwater actuation signal will also be generated at this steam generator water level. Isolation of steam generator blowdown will conserve steam generator water inventory following a loss of main feedwater. The steam generator blowdown isolation valves will also close automatically upon receipt of a containment isolation signal or a high radiation signal (steam generator blowdown or condenser air ejector discharge).

3/4.7.2 ~~STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION~~ DELETED

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

The OPERABILITY of the Reactor Building Closed Cooling Water (RBCCW) System ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

The RBCCW loops are redundant of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a design basis accident, one RBCCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two RBCCW loops must be OPERABLE, and independent to the extent necessary to ensure that a single failure will not result in the unavailability

January 10, 2002
PTSCR 2-18-01

3/4.7.4 SERVICE WATER SYSTEM (Continued)

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the service water pumps differential pressure test, Surveillance Requirement 4.7.4.1.a.2, a substantial flow test, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the service water pumps was developed assuming a 7% degraded pump from the actual pump curves. Flow and pressure measurement instrument inaccuracies for the service water pumps have been accounted for in the design basis hydraulic analysis. It is not necessary to account for flow and pressure measurement instrument inaccuracies in the acceptance criteria contained in the surveillance procedure.

3/4.7.5 ~~FLOOD LEVEL~~ DELETED

~~The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure operator accessibility to the intake structure action to provide pump motor protection will be initiated when the water level reaches plant grade level.~~

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm $\pm$ 10%.

3/4.7.7 ~~SEALED SOURCE CONTAMINATION~~ DELETED

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensureate with the probability of damage to a source in that group. Those sources which are not frequently handled are required to be tested more often thatn those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation mornitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 ~~SNUBBERS~~ DELETED

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guide 8.8 and 8.10. The addition or deletion of any hydraulic or mechanic snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber.

The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber reliability, a representative sample of the installed snubbers will be tested during plant shutdowns at eighteen (18) month intervals. Observed failures of these sample snubbers shall require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.9 DELETED

Docket Nos. 50-336
50-423
B18556

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

Technical Specifications Change Request 2-21-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Retyped Pages

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS

4.4.9.1

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

b. DELETED

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The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. Operation of the RCS within the limits of the heatup and cooldown curves will ensure compliance with this requirement.

Included in this evaluation is consideration of flange protection in accordance with 10 CFR 50, Appendix G. The requirement makes the minimum temperature RT_{NDT} plus 90°F for hydrostatic test and RT_{NDT} plus 120°F for normal operation when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure. Since the flange region RT_{NDT} has been calculated to be 30°F, the minimum flange pressurization temperature during normal operation is 150°F (161°F with instrument uncertainty) when the pressure exceeds 20% of the preservice hydrostatic pressure. Operation of the RCS within the limits of the heatup and cooldown curves will ensure compliance with this requirement.

To establish the minimum boltup temperature, ASME Code Section XI, Appendix G, requires the temperature of the flange and adjacent shell and head regions shall be above the limiting RT_{NDT} temperature for the most limiting material of these regions. The RT_{NDT} temperature for that material is 30°F. Adding 10.5°F, for temperature measurement uncertainty, results in a minimum boltup temperature of 40.5°F. For additional conservatism, a minimum boltup temperature of 70°F is specified on the heatup and cooldown curves. The head and vessel flange region temperature must be greater than 70°F, whenever any reactor vessel stud is tensioned.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements. Verification that pressurizer temperature conditions are within the limits of LCO 3.4.9.2, at least once per 30 minutes, is required when undergoing planned changes of $\geq$ 10°F. The 30 minute time interval permits assessment and correction for temperature deviations within a reasonable time.

a feedwater isolation signal since the steam line break accident analysis credits them in prevention of feed line volume flashing in some cases. Feedwater pumps are assumed to trip immediately with an MSI signal.

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

The atmospheric dump valve (ADV) lines provide a method to maintain the unit in HOT STANDBY, and to replace or supplement the condenser steam dump valves to cool the unit to Shutdown Cooling (SDC) entry conditions. Each ADV line contains an air operated ADV, and an upstream manual isolation valve. The manual isolation valves are normally open, and the ADVs closed. The ADVs, which are normally operated from the main control room, can be operated locally using a manual handwheel.

An ADV line is OPERABLE if local manual operation of the associated valves can be used to perform a controlled release of steam to the atmosphere. This is consistent with the LOCA analysis which credits local manual operation of the ADV lines for accident mitigation.

3/4.7.1.8 STEAM GENERATOR BLOWDOWN ISOLATION VALVES

The steam generator blowdown isolation valves will isolate steam generator blowdown on low steam generator water level. An auxiliary feedwater actuation signal will also be generated at this steam generator water level. Isolation of steam generator blowdown will conserve steam generator water inventory following a loss of main feedwater. The steam generator blowdown isolation valves will also close automatically upon receipt of a containment isolation signal or a high radiation signal (steam generator blowdown or condenser air ejector discharge).

3/4.7.2 DELETED

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

The OPERABILITY of the Reactor Building Closed Cooling Water (RBCCW) System ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

The RBCCW loops are redundant of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a design basis accident, one RBCCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two RBCCW loops must be OPERABLE, and independent to the extent necessary to ensure that a single failure will not result in the unavailability

3/4.7.4 SERVICE WATER SYSTEM (Continued)

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the service water pumps differential pressure test, Surveillance Requirement 4.7.4.1.a.2, a substantial flow test, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the service water pumps was developed assuming a 7% degraded pump from the actual pump curves. Flow and pressure measurement instrument inaccuracies for the service water pumps have been accounted for in the design basis hydraulic analysis. It is not necessary to account for flow and pressure measurement instrument inaccuracies in the acceptance criteria contained in the surveillance procedure.

3/4.7.5 DELETED

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm $\pm$ 10%.

3/4.7.7 DELETED

3/4.7.8 DELETED

3/4.7.9 DELETED

Docket Nos. 50-336
50-423
B18556

Attachment 5

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-18-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Marked Up Pages

List of Affected Pages

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
Index Page viii		Amend. No. 115
Index Page x		Amend. No. 160
Index Page xiv		Amend. No. 136
Index Page xv		Amend. No. 192
3/4.4.9.1	Pressure/Temperature Limits	3/4 4-33, Amend. No. 197 3/4 4-36, Amend. No. 197
3/4.7.2	Steam Generator Pressure/Temperature Limitation	3/4 7-10, Original issue
3/4.7.6	Flood Protection	3/4 7-14, Amend. No. 144
3/4.7.10	Snubbers	3/4 7-22, Amend. No. 71 3/4 7-23, Amend. No. 167 3/4 7-24, Amend. No. 16 3/4 7-25, Original issue 3/4 7-26, Amend. No. 173 3/4 7-27, Amend. No. 100 3/4 7-28, Amend. No. 100 3/4 7-29, Amend. No. 100
3/4.7.11	Sealed Source Contamination	3/4 7-30, Amend. No. 100 3/4 7-31, Amend. No. 100
3/4.7.14	Area Temperature Monitoring	3/4 7-32, Amend. No. 141 3/4 7-33, Amend. No. 100 3/4 7-34, Amend. No. 100 3/4 7-35, Amend. No. 182
3/4.4.9 Bases	Pressure/Temperature Limits	B 3/4 4-12, Amend. No. 197
3/4.7.2 Bases	Steam Generator Pressure/Temperature Limitation	B 3/4 7-7, Amend. No. 151
3/4.7.6 Bases	Flood Protection	B 3/4 7-10, Amend. No. 144
3/4.7.10 Bases, 3/4.7.11 Bases, 3/4.7.14 Bases	Snubbers, Sealed Source Contamination, Area Temperature Monitoring	B 3/4 7-23, Amend. No. 184, B 3/4 7-24, Amend. No. 136, B 3/4 7-25, Amend. No. 136

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

~~August 27, 2001~~

LIMITING CONDITION FOR OPERATION

3.4.9.1 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 ≤ 160°F.

APPLICABILITY: At all times.

ACTION:

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.

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.~~

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

For Information Only

No Change

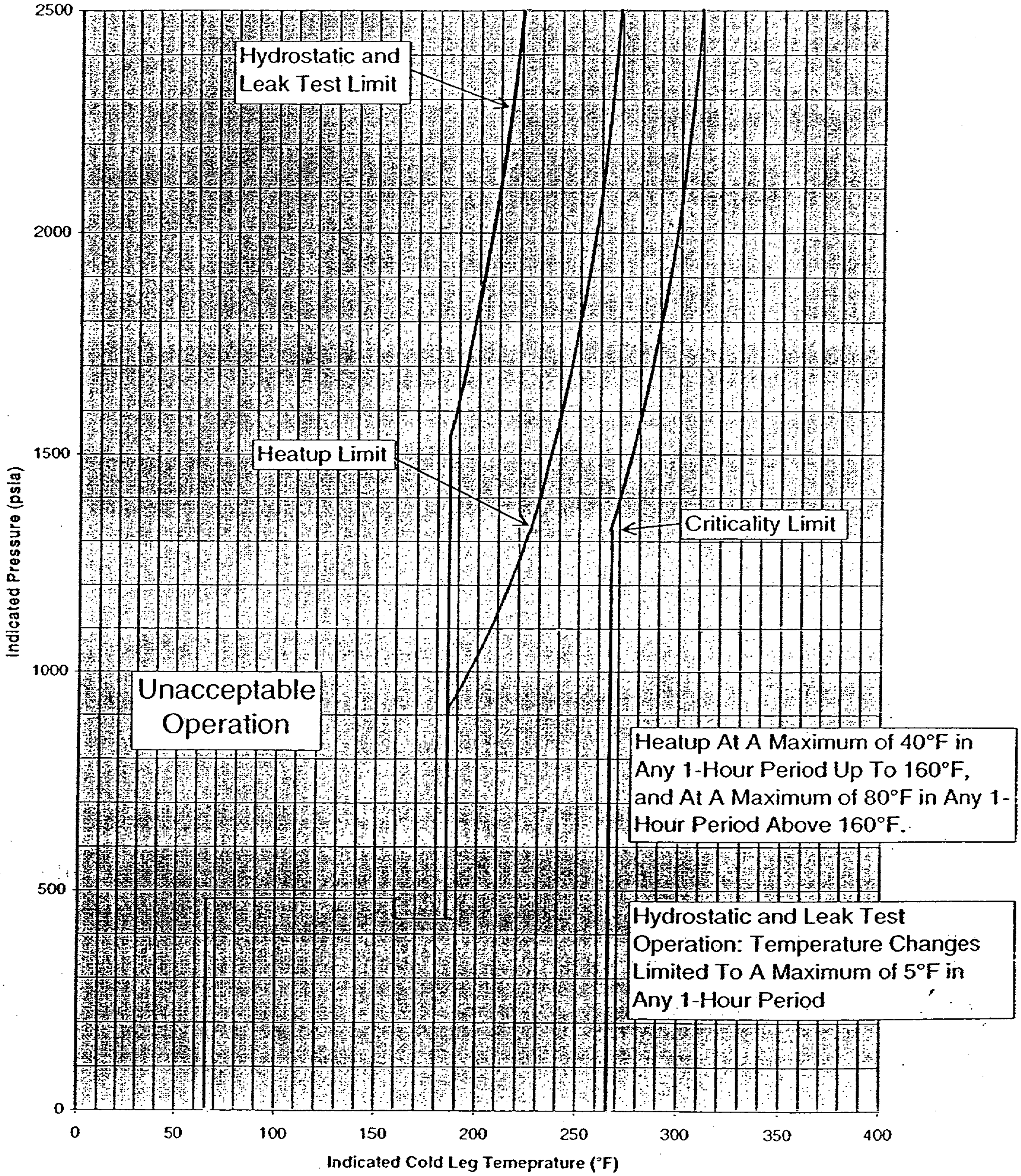

FIGURE 3.4-2

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

For Information Only
No Change

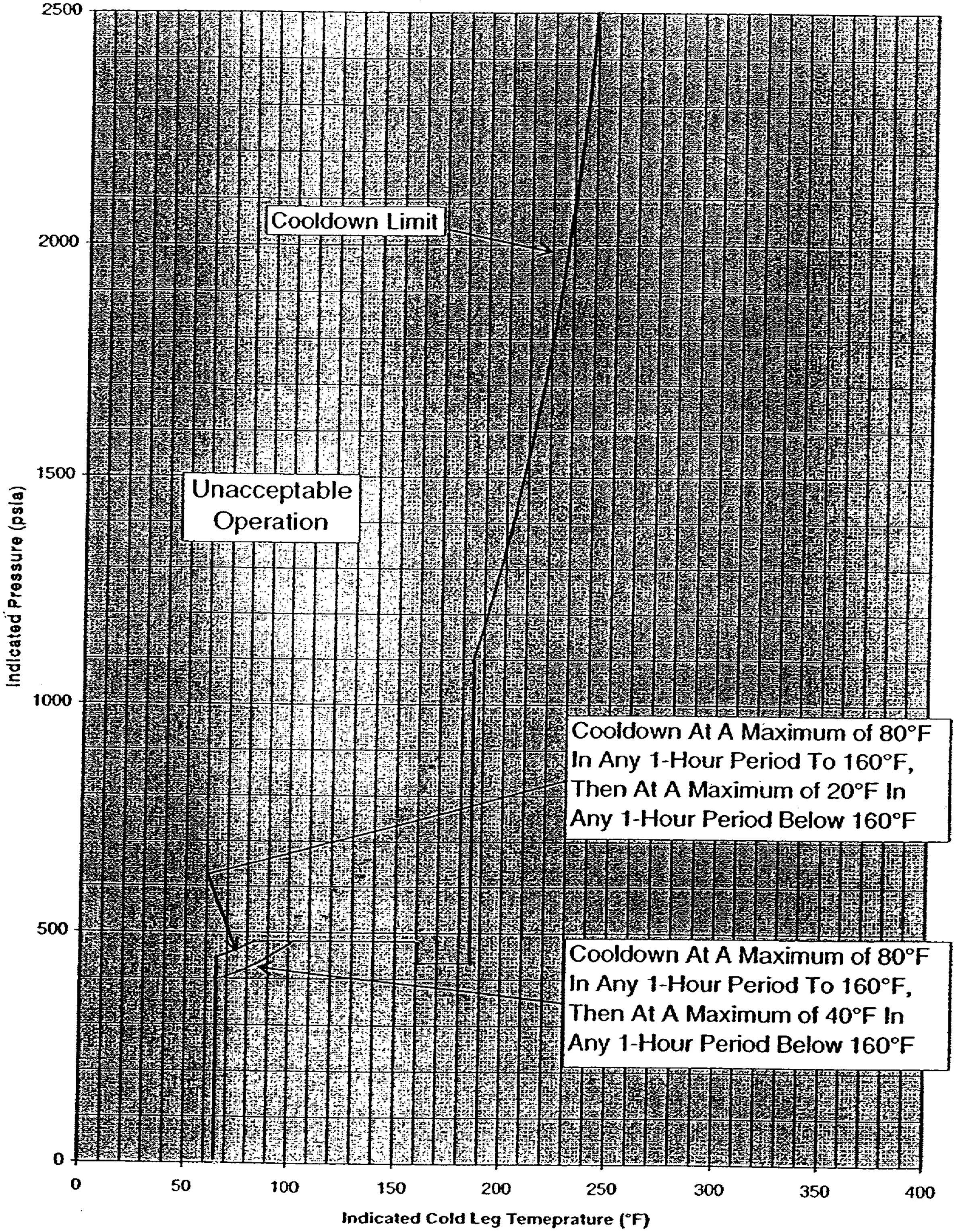

FIGURE 3.4-3

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.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

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~~PLANT SYSTEMS~~

~~3/4.7.6 FLOOD PROTECTION~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.7.6 Flood protection shall be provided for the service water pump cubicles and components when the water level exceeds 13 feet Mean Sea Level, USGS datum, at the Unit 3 intake structure.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~With the water level at greater than 13 feet above Mean Sea Level, USGS datum, at the Unit 3 intake structure, shut the watertight doors of both service water pump cubicles and close the pump cubicle sump drain valves within 15 minutes.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.6 The water level at the Unit 3 intake structure shall be determined to be within the limits by:~~

~~a. Measurement at least once per 24 hours when the water level is below elevation 8 feet above Mean Sea Level, USGS datum, and~~

~~b. Measurement at least once per 2 hours when the water level is equal to or above elevation 8 feet above Mean Sea Level, USGS datum.~~

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3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.10g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-2.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of

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SURVEILLANCE REQUIREMENTS (Continued)

type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.10.f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once each REFUELING INTERVAL thereafter,* a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan for each type shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.10f., an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

*Except the surveillance related to snubber functional testing due no later than March 10, 1999 may be deferred until the end of the next refueling outage or no later than September 10, 1999, whichever is earlier.

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SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (Continued)

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.10f. The cumulative number of snubbers of a type tested is denoted by "N". Test results shall be plotted sequentially in the order of sample assignment (i.e. each snubber shall be plotted by its assigned order in the random sample, not by the order of testing). If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

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SURVEILLANCE REQUIREMENTS (Continued)

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.10e. for snubbers not meeting the functional test acceptance criteria.

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SURVEILLANCE REQUIREMENTS (Continued)

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Quality Assurance Program Topical Report.

~~January 3, 1995~~

TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer included a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but no greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

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~~January 3, 1995~~

TABLE 4.7-2

SNUBBER VISUAL INSPECTION INTERVAL

~~Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.~~

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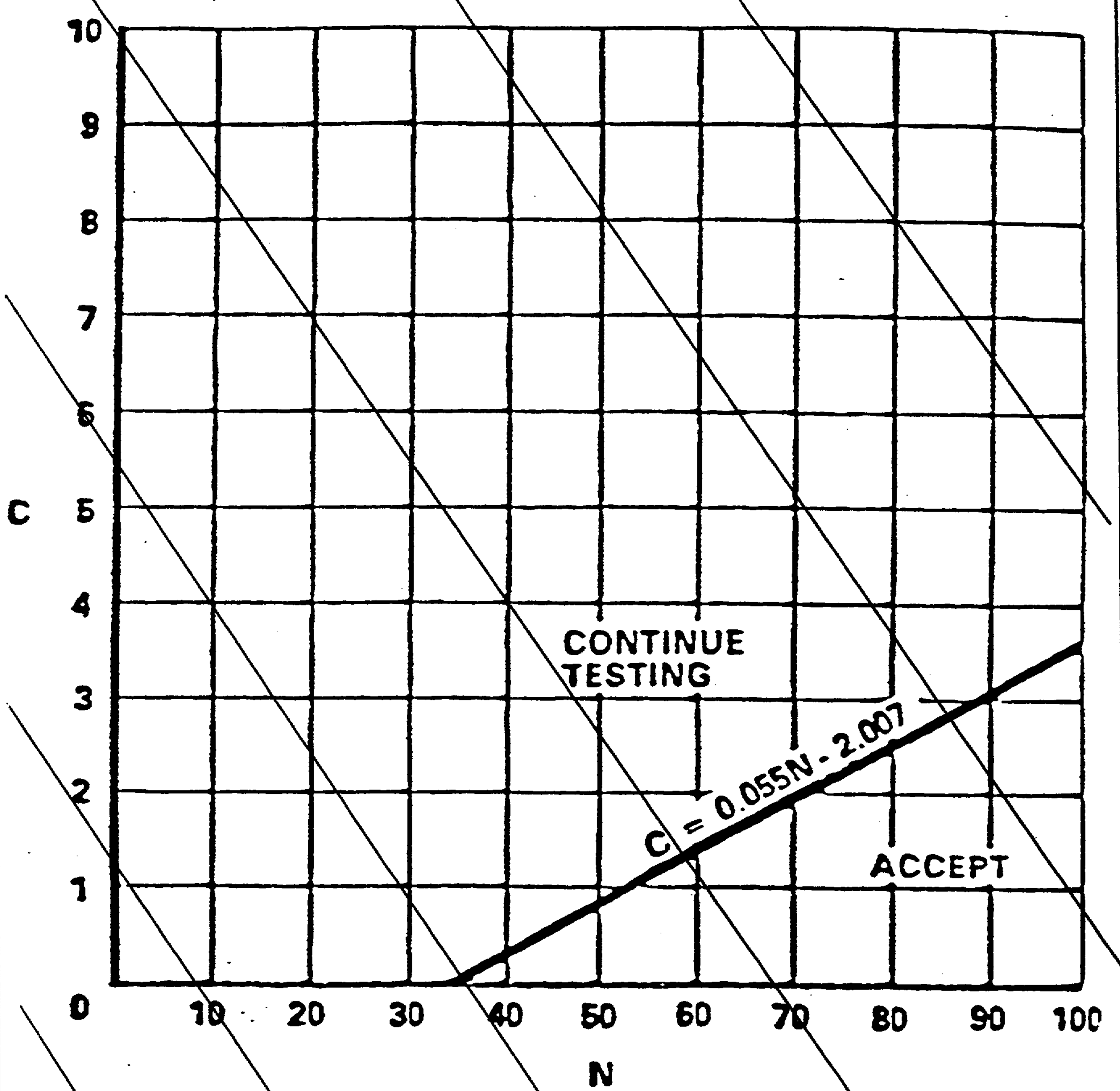

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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PLANT SYSTEMS

3/4.7.11 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.11 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.11.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.11.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

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PLANT SYSTEMS

3/4.7.14 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.14 The temperature limit of each area shown in Table 3.7-6 shall not be exceeded.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6:

a. By less than 20°F and for less than 8 hours, record the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s).

b. By less than 20°F and for greater than or equal to 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specification 3.0.3 are not applicable.

c. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by greater than or equal to 20°F, prepare and submit a Special Report as required by ACTION b. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.14 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limits:

a. At least once per seven days when the alarm is OPERABLE, and;

b. At least once per 12 hours when the alarm is inoperable.

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TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1. AUXILIARY BUILDING	
AB-02, VCT and Boric Acid Transfer Pump Area, El 43′6"	≤ 120
AB-03, Charging Pump Area, El 24′6"	≤ 110
AB-04, General Area, El 66′6"	≤ 120
AB-06, General Area, El 43′6"	≤ 120
AB-07, General Area, El 4′6"	≤ 120
AB-08, General Area (East), El 4′6"	≤ 120
AB-09, General Area (South), El 4′6"	≤ 120
AB-10, General Area, El 4′6"	≤ 120
AB-11, General Area, El 43′6"	≤ 120
AB-13, General Area (North), El 4′6"	≤ 120
AB-16, Supplemental Leak Collection Filter Area, El 66′6"	≤ 120
AB-19, MCC/Rod Drive Area, El 24′6"	≤ 120
AB-21, MCC Air Conditioning Room, El 66′6"	≤ 120
AB-22, Rod Drive Area, El 43′6"	≤ 120
AB-25, Charging Pump Area, El 24′6"	≤ 110
AB-26, RPCCW Pump Area, El 24′6"	≤ 110
AB-29, General Area (Southeast), El 24′6"	≤ 120
AB-33, Boric Acid Tank Area, El 43′6"	≤ 120
AB-35, Boric Acid Tank Area, El 43′6"	≤ 120
AB-39, Fuel Building and Auxiliary Building Filter Area, El 66′6"	≤ 120

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TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA		TEMPERATURE LIMIT (°F)
2.	CONTROL BUILDING	
CB-01,	Switchgear and Battery Rooms, El 4'6"	≤ 104
CB-02,	Cable Spreading Room, El 24'6"	≤ 110
CB-03,	Control and Computer Rooms, El 47'6"	≤ 95
CB-04,	Chiller Room, El 64'6"	≤ 104
CB-05,	Mechanical Equipment Room, El 64-6"	≤ 104
3.	CONTAINMENT	
CS-01,	Inside Crane Wall, El all except CS-03 and CS-04	≤ 120
CS-02,	Outside Crane Wall, El all	≤ 120
CS-03,	Pressurizer Cubicle, El all	≤ 130
CS-04,	Inside Crane wall, El 51'4" except CS-03 and steam generator enclosures	≤ 120
4.	INTAKE STRUCTURE	
CW-01,	Entire Building	≤ 110
5.	DIESEL GENERATOR BUILDING	
DG-01,	Entire Building	≤ 120
6.	ESF BUILDING	
ES-01,	HVAC and MCC Area, El 36'6"	≤ 110
ES-02,	SIH Pump Area, El 21'6"	≤ 110
ES-03,	Pipe Tunnel Area, El 4'6"	≤ 110
ES-04,	RHS Cubicles, El all	≤ 110
ES-05,	RSS Cubicles, El all	≤ 110
ES-06,	Motor Driven Auxiliary Feedwater Pump Area, El 24'-6"	≤ 110
ES-07,	Turbine Driven Auxiliary Feedwater Pump Area, El 24'6"	≤ 110

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TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
7. FUEL BUILDING	
FB-02, Fuel Pool Pump Cubicles, El 24'6"	≤ 119
FB-03, General Area, El 52'4"	≤ 108
8. FUEL OIL VAULT	
FV-01, Diesel Fuel Oil Vault	≤ 95
9. HYDROGEN RECOMBINER BUILDING	
HR-01, Recombiner Skid Area, El 24'6"	≤ 125
HR-02, Controls Area, El 24'6"	≤ 110
HR-03, Sampling Area, El 24'6"	≤ 110
HR-04, HVAC Area, El 37'6"	≤ 110
10. MAIN STEAM VALVE BUILDING	
MS-01, Areas above El. 58'0"	≤ 140
MS-02, Areas below El. 58'0"	≤ 140
11. TURBINE BUILDING	
TB-01, Entire Building	≤ 115
12. TUNNEL	
TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building	≤ 112
13. YARD	
YD-01, Yard	≤ 115

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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.

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.

NO change, For information only.

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.

The LCO establishes operating limits that provide a margin to brittle failure, applicable to the ferritic material 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".

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.

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.

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. 4) and Appendix H of 10 CFR 50 (Ref. 5). 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. 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.

No change, for Information only.

PRESSURE/TEMPERATURE LIMITS (continued)

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 160°F above the minimum permissible temperature for ISLH testing. This limit provides the required margin relative to brittle fracture. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.4, "Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the ferritic RCPB materials, 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. 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 1, as modified by Reference 2, combined with the additional requirements of Reference 3 provide 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.

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

LCO

The LCO limits apply to the ferritic 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 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.

NO change, For Information Only.

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.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure of ferritic RCS components using ASME Section XI, Appendix G, as modified by Code Case N-640 and the additional requirements of 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

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 Allowed Outage Times (AOTs) 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.

NO Change, for Information Only

PRESSURE/TEMPERATURE LIMITS (continued)

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour AOT when operating in MODES 1 through 4 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.

This evaluation must be completed whenever a limit is exceeded. Restoration within the AOT alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

If the required remedial actions are not completed within the allowed times, the plant must be placed in a lower MODE or not allowed to enter MODE 4 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 evaluation for continued operation in MODES 1 through 4 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.

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.

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.

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 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.

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.

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.

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

REFERENCES

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

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.

3/4.7.1.6 STEAM GENERATOR ATMOSPHERIC RELIEF BYPASS LINES

The OPERABILITY of the steam generator atmospheric relief bypass valve (SGARBV) lines provides a method to recover from a steam generator tube rupture (SGTR) event during which the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to limit the primary to secondary break flow into the ruptured steam generator. The time required to limit the primary to secondary break flow for an SGTR event is more critical than the time required to cooldown to RHR entry conditions. Because of these time constraints, these valves and associated flow paths must be OPERABLE from the control room. The number of SGARBVs required to be OPERABLE from the control room to satisfy the SGTR accident analysis requires consideration of single failure criteria. Four SGARBV are required to be OPERABLE to ensure the credited steam release pathways available to conduct a unit cooldown following a SGTR.

For other design events, the SGARBVs provide a safety grade method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the steam bypass system or the steam generator atmospheric relief valves be unavailable. Prior to operator action to cooldown, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below design limits.

Each SGARBV line consists of one SGARBV and an associated block valve (main steam atmospheric relief isolation valve, 3MSS*MOV18A/B/C/D). These block valves are used in the event a steam generator atmospheric relief valve (SGARV) or SGARBV fails to close. Because of the electrical power relationship between the SGARBV and the block valves, if a block valve is maintained closed, the SGARBV flow path is inoperable because of single failure consideration.

The bases for the required actions can be found in NUREG 1431, Rev. 1.

The LCO APPLICABILITY and ACTION statements uses the terms "MODE 4 when steam generator is relied upon for heat removal" and "in MODE 4 without reliance upon steam generator for heat removal." This means that those steam generators which are credited for decay heat removal to comply with LCO 3.4.1.3 (Reactor Coolant System, Hot Shutdown) shall have an OPERABLE SGARBV line. See Bases Section 3/4.4.1 for more detail.

3/4.7.2 ~~STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION~~ DELETED

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

SURVEILLANCE REQUIREMENTS

For the surveillance requirements, the UHS temperature is measured at the locations described in the LCO write-up provided in this section.

Surveillance Requirement 4.7.5.a verifies that the UHS is capable of providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature. The 24-hour frequency is based on operating experience related to trending of the parameter variations during the applicable modes. This surveillance requirement verifies that the average water temperature of the UHS is less than or equal to 75°F.

Surveillance Requirement 4.7.5.b requires that the UHS temperature be monitored on an increased frequency whenever the UHS temperature is greater than 70°F during the applicable modes. The intent of this Surveillance Requirement is to increase the awareness of plant personnel regarding UHS temperature trends above 70°F. The frequency is based on operating experience related to trending of the parameter variations during the applicable modes.

3/4.7.6 ~~FLOOD PROTECTION~~ DELETED

~~The limitation on flood protection ensures that the service water pump cubicle watertight doors will be closed and the pump cubicle sump drain valves will be closed before the water level reaches the critical elevation of 14.5 feet Mean Sea Level. Elevation 14.5 feet MSL is the floor elevation of the service water pump cubicle.~~

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

BACKGROUND

The control room emergency ventilation system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. Additionally, the system provides temperature control for the control room during normal and post-accident operations.

The control room emergency ventilation system is comprised of the control room emergency air filtration system and a temperature control system.

The control room emergency air filtration system consists of two redundant systems that recirculate and filter the control room air. Each control room emergency air filtration system consists of a moisture separator, electric heater, prefilter, upstream high efficiency particulate air (HEPA) filter, charcoal adsorber, downstream HEPA filter, and fan. Additionally, ductwork, valves or dampers, and instrumentation form part of the system.

Normal Operation

A portion of the control room emergency ventilation system is required to operate during normal operations to ensure the temperature of the control room is maintained at or below 95°F.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

The OPERABILITY of the Auxiliary Building Filter System ensures that radioactive materials leaking from the equipment within the charging pump, component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. The charging pump/reactor plant component cooling water pump ventilation system must be operational to ensure operability of the auxiliary building filter system and the supplementary leak collection and release system. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

3/4.7.10 ~~SNUBBERS~~ DELETED

~~All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. For the purpose of declaring the affected system OPERABLE with the inoperable snubber(s), an engineering evaluation may be performed, in accordance with Section 50.59 of 10 CFR Part 50.~~

~~Snubbers are classified and grouped by design and manufacturer but not by size. Snubbers of the same manufacturer but having different internal mechanisms are classified as different types. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.~~

~~A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g.,~~

April 10, 1997

3/4.7.10 SNUBBERS (Continued)

temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on asafety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 5% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted

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PLANT SYSTEMS

BASES

3/4.7.10 SNUBBERS (Continued)

shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.11 ~~SEALED SOURCE CONTAMINATION~~ DELETED

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.14 ~~AREA TEMPERATURE MONITORING~~ DELETED

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±2.2°F.

Docket Nos. 50-336
50-423
B18556

Attachment 6

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-18-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System and Plant Systems
Retyped Pages

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

INDEX

BASES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 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 ≤ 160°F.

APPLICABILITY: At all times.

ACTION:

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.

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.

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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.

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.

REFERENCES

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

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.

3/4.7.1.6 STEAM GENERATOR ATMOSPHERIC RELIEF BYPASS LINES

The OPERABILITY of the steam generator atmospheric relief bypass valve (SGARBV) lines provides a method to recover from a steam generator tube rupture (SGTR) event during which the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to limit the primary to secondary break flow into the ruptured steam generator. The time required to limit the primary to secondary break flow for an SGTR event is more critical than the time required to cooldown to RHR entry conditions. Because of these time constraints, these valves and associated flow paths must be OPERABLE from the control room. The number of SGARBVs required to be OPERABLE from the control room to satisfy the SGTR accident analysis requires consideration of single failure criteria. Four SGARBV are required to be OPERABLE to ensure the credited steam release pathways available to conduct a unit cooldown following a SGTR.

For other design events, the SGARBVs provide a safety grade method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the steam bypass system or the steam generator atmospheric relief valves be unavailable. Prior to operator action to cooldown, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below design limits.

Each SGARBV line consists of one SGARBV and an associated block valve (main steam atmospheric relief isolation valve, 3MSS*MOV18A/B/C/D). These block valves are used in the event a steam generator atmospheric relief valve (SGARV) or SGARBV fails to close. Because of the electrical power relationship between the SGARBV and the block valves, if a block valve is maintained closed, the SGARBV flow path is inoperable because of single failure consideration.

The bases for the required actions can be found in NUREG 1431, Rev. 1.

The LCO APPLICABILITY and ACTION statements uses the terms "MODE 4 when steam generator is relied upon for heat removal" and "in MODE 4 without reliance upon steam generator for heat removal." This means that those steam generators which are credited for decay heat removal to comply with LCO 3.4.1.3 (Reactor Coolant System, Hot Shutdown) shall have an OPERABLE SGARBV line. See Bases Section 3/4.4.1 for more detail.

3/4.7.2 DELETED

SURVEILLANCE REQUIREMENTS

For the surveillance requirements, the UHS temperature is measured at the locations described in the LCO write-up provided in this section.

Surveillance Requirement 4.7.5.a verifies that the UHS is capable of providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature. The 24-hour frequency is based on operating experience related to trending of the parameter variations during the applicable modes. This surveillance requirement verifies that the average water temperature of the UHS is less than or equal to 75°F.

Surveillance Requirement 4.7.5.b requires that the UHS temperature be monitored on an increased frequency whenever the UHS temperature is greater than 70°F during the applicable modes. The intent of this Surveillance Requirement is to increase the awareness of plant personnel regarding UHS temperature trends above 70°F. The frequency is based on operating experience related to trending of the parameter variations during the applicable modes.

3/4.7.6 DELETED

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

BACKGROUND

The control room emergency ventilation system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. Additionally, the system provides temperature control for the control room during normal and post-accident operations.

The control room emergency ventilation system is comprised of the control room emergency air filtration system and a temperature control system.

The control room emergency air filtration system consists of two redundant systems that recirculate and filter the control room air. Each control room emergency air filtration system consists of a moisture separator, electric heater, prefilter, upstream high efficiency particulate air (HEPA) filter, charcoal adsorber, downstream HEPA filter, and fan. Additionally, ductwork, valves or dampers, and instrumentation form part of the system.

Normal Operation

A portion of the control room emergency ventilation system is required to operate during normal operations to ensure the temperature of the control room is maintained at or below 95°F.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

The OPERABILITY of the Auxiliary Building Filter System ensures that radioactive materials leaking from the equipment within the charging pump, component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. The charging pump/reactor plant component cooling water pump ventilation system must be operational to ensure operability of the auxiliary building filter system and the supplementary leak collection and release system. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

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