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

Dominion℠

JUN - 4 2001

Docket Nos. 50-336
50-423
B18288

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 2-2-01 and 3-2-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System

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. DNC is proposing to change Technical Specifications 3.4.7, "Chemistry," 3.4.9.2, "Pressure/Temperature Limits - Pressurizer," and 3.4.11, "Reactor Coolant System Vents" for Millstone Unit Nos. 2 and 3. Index pages vi and xii, Technical Specification 3.4.10, "Structural Integrity," and Technical Specification 6.9, "Special Reports" will be changed for Millstone Unit No. 2. Index pages vii, viii, xiii, xiv and xix 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 to the respective Technical Requirements Manual (TRM), with the exception of Millstone Unit No. 3 Technical Specification Section 4.4.10, which will be relocated to Section 6 of the unit's Technical Specifications. Information which is relocated to the TRM will be maintained in accordance with the provisions of 10 CFR 50.59.

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

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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 reviewed the proposed License Amendment Request against the criteria of 10 CFR 51.22 for environmental considerations. These changes will not increase the type and amounts of effluents that may be released offsite. In addition, this amendment request will not increase individual or cumulative occupational radiation exposures. Therefore, DNC has determined the proposed changes will not have an effect on the quality of the human environment.

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 March 31, 2002, with each amendment to be implemented within 60 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.

Raymond P. Necci
Vice President - Nuclear Operations - Millstone

Sworn to and subscribed before me

this 4th day of June, 2001

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. I. Zimmerman, NRC Project Manager, Unit No. 2
S. R. Jones, Senior Resident Inspector, Unit No. 2
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
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Docket Nos. 50-336
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Attachment 1

Millstone Nuclear Power Station, Unit Nos. 2 and 3

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

Technical Specifications Change Requests 2-2-01 and 3-2-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System
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. DNC is proposing to relocate Technical Specifications 3.4.7, "Chemistry," 3.4.9.2, "Pressure/Temperature Limits - Pressurizer," and 3.4.11, "Reactor Coolant System Vents" to the Technical Requirements Manual (TRM) for each unit. Millstone Unit No. 3 Technical Specification 3.4.10, "Structural Integrity," will be relocated to the facility's TRM with the exception of the associated surveillance requirements for this technical specification (Technical Specification 4.4.10 - Reactor Coolant Pump Flywheel Inspection) which will be relocated to Section 6 of the unit's Technical Specifications.

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 3.4.7

Millstone Unit Nos. 2 and 3 Technical Specification 3.4.7 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 page will be deleted and replaced with, "This page intentionally left blank."

Technical Specification 3.4.9.2

Millstone Unit Nos. 2 and 3 Technical Specification 3.4.9.2 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."

Technical Specification 3.4.10

Millstone Unit No. 3 Technical Specification Limiting Condition for Operation (LCO) 3.4.10 will be relocated to the facility's TRM (with the exception of the associated surveillance requirement) where future changes will be controlled in accordance with 10 CFR 50.59. The associated surveillance requirement, Technical Specification 4.4.10 - Reactor Coolant Pump Flywheel Inspection, will be relocated to Section 6 of the unit's Technical Specifications. Relocation of Technical Specification 4.4.10 to Section 6 of the Technical Specifications is consistent with the Standard Technical Specifications for Westinghouse plants, NUREG-1431, Revision 1.

The text on the corresponding pages will be deleted and replaced with, "This page intentionally left blank."

Technical Specification 3.4.11

Millstone Unit Nos. 2 and 3 Technical Specification 3.4.11 will be relocated to the respective unit'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."

Technical Specification 6.0

Technical Specification 6.17, "Reactor Coolant Pump Flywheel Inspection Program," will be created for Millstone Unit No. 3 to reflect the relocation of Technical Specification Surveillance Requirement 4.4.10.

Millstone Unit No. 2 Technical Specification 6.9.2 summarizes those technical specifications which require special reports be submitted under applicable conditions. Millstone Unit No. 2 Technical Specification 6.9.2.m identifies Technical Specification 3.4.11, Reactor Coolant System Vents, as a technical specification which requires a special report under certain conditions. Since Technical Specification 3.4.11 is proposed for relocation, Millstone Unit No. 2 Technical Specification 6.9.2.m will be deleted. This is an administrative change.

Index Pages

Index pages vi and xii for Millstone Unit No. 2 and index pages vii, viii, xiii and xiv for Millstone Unit No. 3 will be revised to reflect the elimination of Technical Specifications 3.4.7, "Chemistry," 3.4.9.2, "Pressure/Temperature Limits - Pressurizer," and 3.4.11, "Reactor Coolant System Vents." Index page xix for Millstone Unit No. 3 will be revised to reflect the relocation of Technical Specification 3.4.10, "Structural Integrity," to the facility TRM and the relocation of the associated surveillance requirement, Technical Specification 4.4.10, to Section 6 of the unit's Technical Specifications.

Technical Specification Bases

The proposed changes to the Bases for Millstone Unit Nos. 2 and 3 Technical Specifications 3.4.7, 3.4.9.2, and 3.4.11 will delete the text associated with each section and replace the section titles with the word, "DELETED." The proposed change to the Bases for Millstone Unit No. 3 Technical Specification 3.4.10, will delete the text associated with this section and replace the section title with the word, "DELETED." These sections 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 relocated while not adversely affecting plant safety. The criteria, and an evaluation of each technical specification proposed for relocation, are provided below.

Technical Specification 3.4.7

Millstone Unit Nos. 2 and 3 Technical Specification 3.4.7 is proposed to be relocated to the respective facility's TRM. This specification ensures that Reactor Coolant System (RCS) chemistry is maintained within specified limits to ensure that corrosion of the RCS is minimized, and to reduce the potential for RCS leakage or failure due to stress corrosion cracking.

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 RCS chemistry parameters monitored by this specification (oxygen, chloride, and fluoride) are not applicable to installed instrumentation which is used to detect, and indicate in the control room, an abnormal degradation of the reactor coolant pressure boundary. This specification does 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 RCS chemistry parameters which are monitored do not provide direct input to Reactor Protection System or Engineered Safety Features Actuation System functions, nor are the RCS chemistry requirements 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.

RCS chemistry requirements are indirectly applicable to a design feature (RCS integrity) that is an initial condition of a DBA or transient analysis that either assumes the failure or presents a challenge to the integrity of a fission product barrier, but the RCS chemistry requirements are not credited with assuring RCS integrity. RCS integrity is assured through inservice inspection and engineering evaluations of structural integrity. Therefore, this specification does not meet Criterion 2.

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.

The RCS chemistry parameters which are monitored do not provide direct input to Reactor Protection System or Engineered Safety Features Actuation System functions. RCS chemistry requirements are applicable to the integrity of the RCS, which is a system 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. However, the chemistry requirements do not directly assure RCS integrity. The chemistry requirements provide an indication of a concern which could adversely affect RCS integrity if left uncorrected long-term. RCS integrity is assured through inservice inspection and engineering evaluations of structural integrity. Therefore, this specification does not meet Criterion 3.

Criterion 4

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

The RCS chemistry requirements covered by this technical specification have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. Primary system corrosion is a slow process which would be detected in inservice inspections. This technical specification does not cover a SSC requiring risk review/unavailability monitoring. This specification does not meet Criterion 4.

This technical specification does 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, this technical specification can be relocated to the TRM.

Technical Specification 3.4.9.2

Millstone Unit Nos. 2 and 3 Technical Specification 3.4.9.2 is proposed to be relocated to the respective facility's TRM. This specification ensures that the Pressurizer temperature maximum heatup and cooldown rates are maintained within the design criteria assumptions for fatigue analysis as required by the ASME Boiler and Pressure Vessel Code.

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 technical specification is not applicable to installed instrumentation which is used to detect, and indicate in the control room, an abnormal degradation of the reactor coolant pressure boundary. This specification does 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.

This technical specification is not applicable to a process variable or design feature 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. This technical specification is applicable to an operating restriction associated with a Transient Analysis which could in the long-term challenge the integrity of a fission product barrier. However, the integrity of the Pressurizer for this operating restriction is maintained through engineering evaluation of the long-term effects of temperature transients, not through any activities performed by the plant staff during operation. This specification does 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.

This technical specification does cover a SSC that is part of the primary success path 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. However, the passive functional integrity of the Pressurizer is not maintained by activities of the operators during plant operation. RCS integrity is assured through inservice inspection and engineering evaluations of structural integrity. This specification does not meet 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 technical specification 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 technical specification does 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, this technical specification can be relocated to the TRM.

Technical Specification 3.4.10

Millstone Unit No. 3 Technical Specification 3.4.10 (with the exception of the associated surveillance requirement) is proposed to be relocated to the facility's TRM. This specification ensures that structural integrity and operational readiness of applicable SSCs will be maintained at an acceptable level throughout the life of the plant through inservice inspection and testing programs. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a.

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.

Those portions of this specification which are being relocated to the facility TRM are not applicable to installed instrumentation which is used to detect, and indicate in the control room, an abnormal degradation of the reactor coolant pressure boundary. This specification does 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.

Those portions of this specification which are being relocated to the facility TRM 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. While this technical specification imposes an operating restriction regarding pressure boundary integrity, it is not monitored or controlled during plant operation. The assumed integrity of Class 1, 2, and 3 components is assured by means of periodic inspections. Therefore, this specification does 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.

ASME Code Class 1, 2, and 3 components are part of the primary success path and function to mitigate DBAs or transients that either assume the failure of or present a challenge to the integrity/operability of these components are included in the individual specifications that cover these components. However, as stated above, those portions of this specification which are being relocated to the facility TRM address the passive, pressure boundary function of these components. Therefore, this technical specification 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 requirements covered by this technical specification which are being relocated to the TRM have not been shown to be risk significant to public health and safety by either operating experience or probabilistic safety assessment. Failure modes of applicable SSCs would not be identified from the requirements of this technical specification. The requirements of this technical specification do not affect the risk review/unavailability monitoring of applicable SSCs. This specification does not meet Criterion 4.

This technical specification does 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, this technical specification can be relocated to the TRM.

Technical Specification 3.4.11

Millstone Unit Nos. 2 and 3 Technical Specification 3.4.11 is proposed to be relocated to the respective facility's TRM. The RCS vents are provided to exhaust noncondensible gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a loss-of-coolant accident. Their function, capabilities, and testing requirements are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements." However, the operation of RCS vents is not assumed in the safety analysis. The operation of these vents is an operator action after the event has occurred, and is only required when there is indication that natural circulation is not occurring.

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 technical specification is not applicable to installed instrumentation which is used to detect, and indicate in the control room, an abnormal degradation of the reactor coolant pressure boundary. This specification does 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.

This technical specification is 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. This specification does 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.

The RCS vents may be used to assist in creating conditions conducive to natural circulation following a design basis accident, but are not SSCs that are 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. This specification does not meet 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 technical specification 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 technical specification does 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, this technical specification can be relocated to the TRM.

Technical Specification 6.9.2.m

Deletion of Millstone Unit No. 2 Technical Specification 6.9.2.m is an administrative change. This change is consistent with the changes previously discussed. Therefore, the proposed changes will have no adverse effect on plant safety.

Index Pages

Revision of Index Pages vi and xii for Millstone Unit No. 2 and Index Pages vii, viii, xiii, xiv, and xix 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 technical specifications will not be modified as a result of the proposed technical specification 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 technical specifications 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.

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Attachment 2

Millstone Nuclear Power Station, Unit Nos. 2 and 3

Technical Specifications Change Requests 2-2-01 and 3-2-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System
<u>Significant Hazards Consideration</u>

Technical Specifications Change Requests 2-2-01 and 3-2-01
Relocation of Selected Technical Specifications Related
to the Reactor Coolant System
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. DNC is proposing to relocate Technical Specifications 3.4.7, "Chemistry," 3.4.9.2, "Pressure/Temperature Limits - Pressurizer," and 3.4.11, "Reactor Coolant System Vents" to the Technical Requirements Manual (TRM) for each unit. Millstone Unit No. 3 Technical Specification 3.4.10, "Structural Integrity," will be relocated to the facility's TRM with the exception of the associated surveillance requirements (Technical Specification 4.4.10 - Reactor Coolant Pump Flywheel Inspection) which will be relocated to Section 6 of the unit's Technical Specifications.

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 to relocate the requirements for reactor chemistry, Pressurizer pressure and temperature limits, structural integrity (with the exception of the Reactor Coolant Pump Flywheel Inspection Program), and Reactor Coolant System (RCS) vent operability from the Technical Specifications to the TRM will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. Therefore, the relocation of the requirements associated with reactor chemistry, Pressurizer pressure and temperature limits, structural integrity (with the exception of the Reactor Coolant Pump Flywheel Inspection Program), and RCS vent operability will not adversely 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 to relocate the requirements for reactor chemistry, Pressurizer pressure and temperature limits, structural integrity (with the exception of the Reactor Coolant Pump Flywheel Inspection Program), and RCS vent operability from the Technical Specifications to the TRM do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. These 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 the requirements for reactor chemistry, Pressurizer pressure and temperature limits, structural integrity (with the exception of the Reactor Coolant Pump Flywheel Inspection Program), and RCS vent operability from the Technical Specifications to the TRM. Any future changes to the relocated requirements will be in accordance with 10 CFR 50.59. The proposed changes will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the DBAs will not change. In addition, the relocated requirements do not meet any of the 10 CFR 50.36c(2)(ii) criteria on items for which technical specifications must be established. Therefore, there will be no reduction in a margin of safety.

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Attachment 3

Millstone Nuclear Power Station, Unit No. 2

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

List of Affected Pages

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
3.4.7	RCS - Chemistry	3/4 4-10, Original Issue 3/4 4-11, Original Issue 3/4 4-12, Original Issue
3.4.9.2	RCS - Pressurizer	3/4 4-21, Amend. No. 218
3.4.11	RCS - RCS Vents	3/4 4-23, Amend. No. 230
Index Page VI		Amend. No. 217
Index Page XII		Amend. No. 217

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

SECTION — PAGE

REACTOR COOLANT SYSTEM

CHEMISTRY replace with "this page intentionally left blank"

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: ALL MODES.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in COLD SHUTDOWN within 36 hours.

MODES 5 and 6

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to ≤ 500 psia, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-1.

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TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN	$\leq$ 0.10 ppm*	$\leq$ 1.00 ppm*
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

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TABLE 4.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
DISSOLVED OXYGEN	3 times per 7 days*	72 hours
CHLORIDE	3 times per 7 days	72 hours
FLUORIDE	3 times per 7 days	72 hours

*Not required with $T_{avg} \leq 250°F$

~~July 1, 1998~~

REACTOR COOLANT SYSTEM

PRESSURIZER

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LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 350°F.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

With any of the above limits exceeded, perform the following:

a. Restore the temperature to within limit within 30 minutes.

AND

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

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperature and spray water temperature differential shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

 Replace with "This page intentionally left blank" ~~March 11, 1999~~

REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series capable of being powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor Vessel head

b. Pressurizer steam space

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

ACTION:

a. With the Pressurizer vent path inoperable, STARTUP and/or POWER OPERATION may continue provided that i) the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path and ii) one power operated relief valve (PORV) and its associated block valve is OPERABLE; otherwise, restore either the inoperable vent path or one PORV and its associated block valve to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the path to OPERABLE status.

b. With the Reactor Vessel Head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided that the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the Reactor Vessel Head vent path to OPERABLE status within 30 days or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the path to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent path through at least once complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during COLD SHUTDOWN or REFUELING.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the REMODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the REMODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary to a dose rate which, if the release were to occur for a full year, would cause a dose of 500 mrem. This conforms to the dose associated with the 1993 version of 10 CFR 20, Appendix B, Table II, Column I;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

6.21 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provided (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the REMODCM, (2) conform to that guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the REMODCM.

3/4.4.7 ~~CHEMISTRY~~ DELETED

~~The limitations on Reactor Coolant System contaminants ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the concentrations of the contaminants within the Steady State Limits shown on Table 3.4-1 provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.~~

~~The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.~~

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 uCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

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.

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.

For information only

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 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.

3/4.4.11 ~~Reactor Coolant System Vents~~ DELETED

~~Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.~~

~~The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.~~

~~The flow test verifies that each flowpath through the two solenoid valves is OPERABLE. This verification can be performed by using a series of overlapping tests to ensure flow is verified through all parts of the system.~~

Docket Nos. 50-336
50-423
B18288

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

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

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

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3/4.4.7 DELETED

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 uCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

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.

3/4.4.11 DELETED

Docket Nos. 50-336
50-423
B18288

Attachment 5

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-2-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System
Marked Up Pages

List of Affected Pages

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
3.4.7	RCS - Chemistry	3/4 4-25, Original Issue 3/4 4-26, Original Issue 3/4 4-27, Original Issue
3.4.9.2	RCS - Pressurizer	3/4 4-37, Amend. No. 157
3.4.10	RCS - Structural Integrity	3/4 4-42, Amend. No. 169
3.4.11	RCS - RCS Vents	3/4 4-43, Amend. No. 100 3/4 4-43a, Amend. No. 133
6-26	Administrative Controls	Amend. No. 188
Index Page vii		Amend. No. 188
Index Page viii		Amend. No. 115
Index Page xiii		Amend. No. 164
Index Page xiv		Amend. No. 136
Index Page xix		Amend. No. 188

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

~~May 26, 1995~~

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

SECTION

ADMINISTRATIVE CONTROLS

REACTOR COOLANT SYSTEM

~~JAN 31 1986~~

3/4.4.7 CHEMISTRY

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LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

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JAN 31 1986

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	< 0.10 ppm	≤ 1.00 ppm
Chloride	< 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

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~~JAN 31 1986~~

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

2/12/98

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REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.~~

~~APPLICABILITY: All MODES.~~

~~ACTION:~~

~~a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.~~

~~b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.~~

~~c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic partical testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.~~

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3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of two parallel trains with two valves inseries powered from emergency busses shall be OPERABLE and the vent closed* at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

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

ACTION:

a. With one train of the reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable train is maintained closed with power removed from the valve actuators of all valves in the inoperable train; restore the inoperable train to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both trains of the reactor vessel head vent paths inoperable; maintain both trains closed with power removed from the valve actuators of all valves in the inoperable trains, and restore at least one of the trains to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any valve(s) of the pressurizer steam space vent path inoperable in MODES 1, 2, or 3, follow the ACTION requirements of Specification 3.4.4.

d. With any valve(s) of the pressurizer steam space vent path inoperable in MODE 4, follow the ACTION requirements of Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each train of the reactor vessel head vent path isolation valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per COLD SHUTDOWN, if not performed within the previous 92 days, by operating the valve through one complete cycle of full travel from the control room.

*For an OPERABLE vent path using a power-operated relief valve (PORV) as the vent path, the PORV block valve is not required to be closed.

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SURVEILLANCE REQUIREMENT (Continued)

~~4.4.11.2 Each train of the reactor vessel head vent path shall be demonstrated OPERABLE at least once each REFUELING INTERVAL by:~~

~~a. Verifying all manual isolation valves in each vent path are locked in the open position,~~

~~b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and~~

~~c. Verifying flow through the Reactor Coolant System vent paths during venting.~~

~~4.4.11.3 Each train of the pressurizer steam space vent path shall be demonstrated OPERABLE per the applicable requirement of Specifications 4.4.4.1 through 4.4.4.3 and 4.4.9.3.1. In addition, flow shall be verified through the pressurizer steam space vent path during venting at least once each REFUELING INTERVAL.~~

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

6.16 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provided (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the REMODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the REMODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

Insert A

Insert A to MP3 page 6-26

6.17 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic partical testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.

OPERATIONAL LEAKAGE (Continued)

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 ~~CHEMISTRY~~ DELETED

~~The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.~~

~~The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.~~

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values

PRESSURE/TEMPERATURE LIMITS (continued)

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

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

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

REFERENCES

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

~~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.~~

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PRESSURIZER (continued)

The LCO contains the Pressurizer limits for heatup, cooldown, and spray water temperature differential. Each temperature limit defines an acceptable region for normal operation. The limits that apply to the Pressurizer are as follows: The Pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the Pressurizer and the spray fluid is greater than 320°F.

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

The consequence of violating the LCO limits is that the Pressurizer has been operated under conditions that can result in failure, 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 Pressurizer.

APPLICABLE SAFETY ANALYSIS

The Pressurizer temperature limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering temperature and temperature rate of change conditions that might cause the initiation/propagation of undetected cracks and cause failure of the pressure boundary.

LCO

The two elements of this LCO are:

a. Limits on the rate of change of temperature; and

b. Limits on the spray water differential temperature.

The LCO limits apply to the Pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the Pressurizer wall and, therefore, restricts stresses caused by thermal gradients.

Violating the LCO limits places the Pressurizer outside of the bounds of the stress analyses. The consequences depend on several factors, as follow:

a. The severity of the rate of change of temperature;

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PRESSURIZER (continued)

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

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

APPLICABILITY

The Pressurizer temperature limits LCO provides a definition of acceptable operation for prevention of failure. The temperature limits were developed to provide requirements for operation during heatup or cooldown, and their Applicability is at all times in keeping with the concern for failure.

ACTIONS

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

Besides restoring operation within limits, an evaluation is required to determine if Pressurizer operation can continue. The evaluation must verify the Pressurizer 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.

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

This evaluation must be completed whenever a limit is exceeded. Restoration within 30 minutes alone is insufficient because higher than analyzed stresses may have occurred and may have affected the Pressurizer integrity.

If the required remedial actions are not completed within the allowed times, the plant must be placed in a lower MODE because a sufficiently severe event may have caused entry into an unacceptable region. This possibility indicates a need for more careful examination of the event, best accomplished with the Pressurizer at reduced pressure. In reduced pressure conditions, the possibility of propagation with undetected flaws is decreased.

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

PRESSURIZER (continued)

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

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

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

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 50.55a(g).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 80 Edition and Addenda through Winter.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function. The reactor vessel head vent path consists of two parallel flow paths with redundant isolation valves (3RCS*SV8095A, 3RCS*SV8096A and 3RCS*SV8095B, 3RCS*SV8096B) in each flow path. The pressurizer steam space vent path consists of two parallel paths with a power operated relief valve (PORV) and PORV block valve in series (3RCS*PCV455A, 3RCS*MV800A and 3RCS*PCV456, 3RCS*MV8000B).

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

<u>Docket Nos. 50-336</u>
<u>50-423</u>
<u>B18288</u>

Attachment 6

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-2-01
Relocation of Selected Technical Specifications
Related to the Reactor Coolant System
<u>Retyped Pages</u>

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

ADMINISTRATIVE CONTROLS

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j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

6.16 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provided (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the REMODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the REMODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

c. Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.17 REACTOR COOLANT PUMP FLYWHEEL INSPECTION PROGRAM

This program shall provide for the inspection of each reactor coolant pump flywheel by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic partical testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.

OPERATIONAL LEAKAGE (Continued)

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 DELETED

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values

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.

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OVERPRESSURE PROTECTION SYSTEMS

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