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

Dominion

APR 26 2001

Docket No. 50-336
B18389

RE: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Millstone Unit No. 2 Technical Specifications.

The proposed changes will add Section 6.22, "Reactor Coolant Pump Flywheel Inspection Program" to Section 6, "Administrative Controls" of the Technical Specifications and relocate the requirements of Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" to the Millstone Unit No. 2 Technical Requirements Manual (TRM). The Bases of the affected Technical Specification will also be relocated to the TRM. The Index pages will be updated to reflect the proposed changes.

Regulatory Guide 1.14,[1] Position C.4.b, requires flywheel inspections at approximately 3-year and 10-year intervals. In particular, the 10-year inspection requires a surface examination of all exposed surfaces and a complete ultrasonic test (UT) volumetric examination. The proposed changes will allow the inspection interval of the Millstone Unit No. 2 Reactor Coolant Pumps (RCP) flywheels to be changed to at least once every 10 years and will limit flywheel examination to the volume from the inner bore of the flywheel to one-half the outer radius for in-place ultrasonic examination, or a surface examination of the exposed surfaces of the flywheel when disassembled. The proposed changes, which are based on Combustion Engineering Owners Group

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(1) Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, 1975.

(CEOG) Topical Report SIR-94-080-A,[2] will eliminate the 3-year inspection and require either a UT examination or a surface examination of the exposed surfaces every 10 years. The proposed changes will reduce personnel radiation exposure and will also reduce inspection costs.

Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the marked-up version of the appropriate pages of the current Technical Specifications. Attachment 4 provides the retyped pages of the Technical Specifications.

## Environmental Considerations

DNC has reviewed the proposed License Amendment Request against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes add Section 6.22 to the administrative controls section of the Technical Specifications and will relocate the requirements of Technical Specification 3/4.4.10 to the TRM. 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 a significant effect on the quality of the human environment.

## Conclusions

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

The Millstone Unit No. 2 Third 10-Year Inservice Inspection (ISI) interval began on April 1, 1999. In accordance with the existing requirements provided under Regulatory Guide 1.14, Position C.4.b, an approximately 3-year inspection is to be completed by performing an in-place UT examination of the areas of higher stress concentration at the bore and keyway coinciding with the ISI schedule as required by Section XI of the ASME Code. According to this required schedule the RCP flywheel 3-year inspection shall be completed by the end of the first inspection period ending on July 31, 2002. In order to meet these existing requirements the next scheduled 3-year inspection of the RCP flywheels is scheduled during Millstone Unit No. 2 refueling outage 14 (currently

[2] Entergy Operations, Inc., letter from J. W. Yelverton (Entergy) to the Nuclear Regulatory Commission with enclosed report, SIR-94-080-A, "Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements," dated April 4, 1995.

scheduled in early February of 2002). Approval of this amendment will allow the inspection interval of the Millstone Unit No. 2 RCP flywheels to be increased to at least once every 10-years and eliminate the currently required 3-year inspection. Therefore, DNC requests issuance of this amendment prior to December 1, 2001, with the 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 regarding this submittal, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

Raymond P. Necci

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 26th day of April, 2001

Sandra J. Anton
Notary Public

Date Commission Expires: SANDRA J. ANTON NOTARY PUBLIC COMMISSION EXPIRES MAY 31, 2005

Attachments (5)

cc: H. J. Miller, Region I Administrator
D. S. Collins, NRC Project Manager, Millstone Unit No. 2
S. R. Jones, Senior Resident Inspector, Millstone Unit No. 2

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
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Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)
Discussion of Changes and Safety Summary

**Proposed Revision to Technical Specifications**
**Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)**
**Discussion of Changes and Safety Summary**

Introduction

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Millstone Unit No. 2 Technical Specifications. The proposed changes will introduce changes to Reactor Coolant Pump (RCP) flywheel inspection requirements by adding Section 6.22, "Reactor Coolant Pump Flywheel Inspection Program" to Section 6, "Administrative Controls" of the Technical Specifications and will relocate the requirements of Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" to the Millstone Unit No. 2 Technical Requirements Manual (TRM). The Bases of the affected Technical Specification will also be relocated to the TRM. The Index pages will be updated to reflect the proposed changes.

Discussion of Technical Specifications Changes

Each proposed change is discussed below:

1. Adding Section 6.22, to the administrative controls section of the Technical Specifications

   Background

   Regulatory Position C.4.b of Regulatory Guide (RG) 1.14[(1)] requires flywheel inspections at approximately 3-year and 10-year intervals. In particular, the 10-year inspection requires a surface examination of all exposed surfaces and a complete ultrasonic test (UT) volumetric examination. Extensive work completed by Structural Integrity Associates for the Combustion Engineering Owners Group (CEOG) and documented in Topical Report SIR-94-080-A[(2)] provided alternatives to the current inspection requirements for reactor coolant pump flywheels. This report is applicable to Millstone Unit No. 2, which is a Combustion Engineering plant. The NRC approved this Topical Report subject to certain plant-specific conditions which apply to Millstone Unit No. 2 under Attachment 6.1 of an NRC Safety Evaluation Report[(3)] (SER).

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(1) Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, 1975.

(2) Entergy Operations, Inc., letter from J. W. Yelverton (Entergy) to the Nuclear Regulatory Commission with enclosed report, SIR-94-080-A, "Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements," dated April 4, 1995.

(3) B. W. Sheron (NRC) to D. C. Mims (Entergy), "Acceptance for Referencing of Topical Report SIR-94-080-A, Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements," dated May 21, 1997.

According to the NRCs' SER, Millstone Unit No. 2 needs to fulfill two requirements to submit a plant-specific application for changes to RCP flywheel inspection requirements. The two requirements are: (1) verify the reference temperature $RT_{NDT}$ for its RCP flywheels, and (2) for the four flywheels made of ASTM-A 516, Grade 70 justify the use of the $K_{Ic}$ v.s. ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code to derive their respective $K_{Ic}$ values.

Description of the Change

Technical Specification Section 6.22, "Reactor Coolant Pump Flywheel Inspection Program," will be added to the administrative section of Millstone Unit No. 2 Technical Specifications. The added section will state that: "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 particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years." This section is added to replace the portion of Surveillance Requirements 4.4.10 which will not be relocated to the TRM as described in the next section below. The portion which will not be relocated states that "each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975."

Regulatory Position C.4.b of Regulatory Guide 1.14 requires flywheel inspections at approximately 3-year and 10-year intervals. The proposed changes will eliminate the 3-year inspection and require either a UT examination or a surface examination of the exposed surfaces every 10 years. The proposed changes will reduce personnel radiation exposure and will also reduce inspection costs.

DNC performed a detailed technical evaluation, which concluded that Millstone Unit No. 2 meets the requirements prescribed in the NRC SER for the Combustion Engineering Owners Group Topical report SIR-94-080-A. Therefore, this proposed change is in accordance with the NRC approved topical report previously mentioned. The technical evaluation of these plant-specific conditional requirements is provided in Attachment 5.

2. Relocation of Technical Specification 3/4.4.10 to TRM

The Millstone Unit No. 2 TRM includes information which has been relocated from Technical Specifications or material which has been judged to warrant administrative control. The TRM is referenced by the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). Modifications to the TRM are performed pursuant to the provisions of 10 CFR 50.59.

The purpose of Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" is to specify the requirements of maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. This specification ensures the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. This specification also delineates the surveillance requirements for RCP flywheels. The requirements of Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" including Limiting Condition for Operation, Applicability, and portion of the Surveillance Requirements will be relocated to the Millstone Unit No. 2 TRM. The relocated portion of the Surveillance Requirement will state that: "The structural integrity of ASME Code Class 1, 2 and 3 components shall be inspected in accordance with the requirements delineated in Specification 4.0.5." The remaining portion of the Surveillance Requirements, which states that "each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975," will be replaced with a description of a Reactor Coolant Pump (RCP) program that will be added to the administrative section of the Technical Specification. The details regarding the added section are discussed separately in item number 1 above. The text on the Technical Specification page will be deleted and replaced with, "This page intentionally left blank."

The portion of this specification which will be relocated to the TRM (whole specification except the portion specifying surveillance requirement for the RCP flywheel) addresses the passive, pressure boundary function of ASME Code Class 1, 2, and 3 components. The portion of the Technical Specification, which will be relocated to the TRM, 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. Detailed discussion of the applicability of 10 CFR 50.36c(2)(ii) criteria is provided below in the safety summary section.

Relocation of Technical Specification 3/4.4.10 and the associated Bases section to the TRM does not imply any reduction in its importance in specifying the requirements of maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. Future changes, after relocation to the TRM, will be controlled in accordance with 10 CFR 50.59.

3. Bases change

The proposed changes to the Bases for Millstone Unit No. 2 Technical Specification 3.4.10 will delete the text associated with this section and replace the section title with the word, "DELETED." This section will be relocated to the TRM.

4. Index changes

The Index pages will be updated to reflect the proposed changes. These changes are administrative in nature.

Safety Summary

1. Adding Section 6.22, to the administrative controls section of the Technical Specifications

   The RCP flywheel inspection requirement will be changed from being inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, to being inspected in accordance with the program delineated in the newly created Section 6.22. This change provides an appropriate level of defense in depth as was concluded in the NRC SER and thus no corresponding reductions in quality or safety will be introduced. Therefore, the proposed changes will have no adverse effect on plant safety.

2. Relocation of Technical Specification 3/4.4.10 and its Bases to the TRM

   The purpose of Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" is to specify the requirements for maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. This specification ensures the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. This specification also delineates the surveillance requirements for RCP flywheels. This specification will be relocated to the TRM except for the portion of the surveillance requirement stating that "each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975." The portion of this specification which will be relocated to the TRM addresses the passive, pressure boundary function of ASME Code Class 1, 2, and 3 components.

   The criteria set forth in the final Commission Policy Statement, which was used to develop the Standard Technical Specifications (STS), have been codified in NRC Regulation 10 CFR 50.36(c)(2)(ii). These criteria are:

   10 CFR 50.36(c)(2)(ii)(A) Criterion 1 states that Technical Specification limiting conditions for operation must be established for "installed instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary."

   10 CFR 50.36(c)(2)(ii)(B) Criterion 2 states that Technical Specification limiting conditions for operation must be established for "process variables that are initial conditions of a design basis accident (DBA) or transient analysis that assume the failure of or present a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses and which are monitored and controlled during power operation.

   10 CFR 50.36(c)(2)(ii)(C) Criterion 3 states that Technical Specification limiting conditions for operation must be established for Structures, Systems, or

Components (SSC) that are part of the primary success path and which function or actuate 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 intent of this criterion is to capture into Technical Specifications only those SSCs that are part of the primary success path of a safety analysis sequence. Also captured in this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety analysis sequence consist of the combination and sequences of equipment needed to operate (including consideration for single failure), so that the plant response to DBAs and transients limits the consequences of the events to within the appropriate acceptance criteria.

10 CFR 50.36(c)(2)(ii)(D) Criterion 4 states that Technical Specification limiting conditions for operation must be established for SSCs which operating experience or probabilistic risk assessment has shown to be significant to the public health and safety. The intent of this criterion is that risk insights and operating experience be factored into the establishment of Technical Specifications limiting conditions for operation. This criterion was developed to cover those insights not fully recognized in the safety analysis report DBA or transient analyses. These insights are used to verify that none of the LCOs to be relocated or eliminated from Technical Specifications contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

The following discussion will show that 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:

<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.

The portion of this specification which is being relocated to the facility TRM is not applicable to installed instrumentation which is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. This specification does not meet 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.

The portion of this specification which is being relocated to the facility TRM 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. 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, which 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, the portion of this specification which is being relocated to the facility TRM addresses 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[(4)]. 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.

Relocation of Technical Specification 3/4.4.10 and the associated Bases section to the TRM does not imply any reduction in its importance in specifying the requirements of maintaining the structural integrity of ASME Code Class 1, 2, and 3 components. Future changes, after relocation to the TRM, will be controlled in accordance with 10 CFR 50.59. Therefore, the proposed changes will have no adverse effect on plant safety.

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(4) Millstone Station, Functional Administrative Procedure, "Conduct of on-line Maintenance," MP-20-WM-FAP02.1, Rev. 0.

3. Index changes

   The Index pages will be updated to reflect the proposed changes. These changes are administrative in nature. Therefore, the proposed changes will have no adverse effect on plant safety.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)
Significant Hazards Consideration

**Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)
Significant Hazards Consideration**

Description of The License Amendment Request

The proposed license amendment request will add Section 6.22, "Reactor Coolant Pump Flywheel Inspection Program" to Section 6, "Administrative Controls" of the Technical Specifications and relocate the requirements of Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" to the Millstone Unit No. 2 Technical Requirements Manual (TRM). The Bases of the affected Technical Specification will also be relocated to the TRM. The Index pages will be updated to reflect the proposed changes.

Significant Hazards Consideration

In accordance with 10 CFR 50.92, Dominion Nuclear Connecticut, Inc. (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 a SHC because the changes would not:

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

   Missile generation from a Reactor Coolant Pump (RCP) flywheel could damage the Reactor Coolant System, the Containment, or other equipment or systems important to safety. The fracture mechanics analyses conducted to support the change to Inservice Inspection (ISI) requirements in accordance with the proposed Section 6.22, "Reactor Coolant Pump Flywheel Inspection Program" shows that a pre-existing crack sized just below the detection level will not grow to the flaw size necessary to create flywheel missiles within the life of the plant. This analysis conservatively assumes minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area, and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in a Millstone Unit No. 2 flywheel will not grow to the allowable flaw size under Loss of Coolant Accident (LOCA) conditions over the life of the plant, reducing the ISI requirements for the detection of such cracks over the life of the plant will not significantly increase the probability or consequences of an accident previously evaluated.

   The proposed Technical Specification changes to relocate the requirements for Technical Specification 3/4.4.10, "Reactor Coolant System, Structural Integrity" (with the exception of the RCP inspection requirements) to the TRM will have no adverse effect on plant operation or the availability or operation of any accident mitigation equipment. Therefore, the Reactor Coolant System structural integrity (with the exception of the RCP flywheel which is addressed above) will not adversely impact an accident initiator and can not cause an accident. Therefore these changes will not increase the probability or consequences of an accident previously evaluated.

The Index pages will be updated to reflect the proposed changes. These changes are administrative in nature. 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 changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These 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 fracture mechanics analyses conducted to support the change to ISI requirements in accordance with the proposed Section 6.22, "Reactor Coolant Pump Flywheel Inspection Program" shows that significant conservatism has been used for calculating the allowable flaw size, critical flaw size, and crack growth rate in the RCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in a Millstone Unit No. 2 flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under LOCA conditions over the life of the plant, reducing ISI requirements for the detection of such cracks over the life of the plant will not involve a significant reduction in the margin of safety. The proposed changes have no impact on plant equipment operation. Therefore, the proposed changes will not result in a reduction in a margin of safety.

Relocation of Technical Specification 3/4.4.10 (whole specification except the portion specifying surveillance requirement for the RCP flywheel) to the TRM does not imply any reduction in its importance in ensuring that the structural integrity and operational readiness of ASME Code Class 1, 2, and 3 components will be maintained at an acceptable level throughout the life of the plant. The proposed change has no impact on plant equipment operation. Therefore, the proposed change will not result in a reduction in a margin of safety.

Docket No. 50-336
B18389

Attachment 3

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)
Marked Up Pages

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

# INDEX

## BASES

REACTOR COOLANT SYSTEM

STRUCTURAL INTEGRITY

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) to within its limit or isolate the affects 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 to within its limit or isolate the affected component from service.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the inspection requirements for ASME Code Classes 1, 2, and 3 components delineated in Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

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b. A Land Use Census to ensure that changes in the use of areas at and beyon.. 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

6.22 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 particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.

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~~ DELETED

~~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.~~

Docket No. 50-336
B18389

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)
Retyped Pages

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

## ADMINISTRATIVE CONTROLS

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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.

6.22 Reactor Coolant Pump Flywheel Inspection Report

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 particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.

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 DELETED

Attachment 5

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Changes to Reactor Coolant Pump Flywheel Inspection Requirements and Relocation of the Reactor Coolant System's Structural Integrity Technical Specification to the Technical Requirements Manual (TSCR 2-7-01)
Evaluation of Plant-Specific Conditional Requirements

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**Evaluation of Plant-Specific Conditional Requirements**

Introduction

According to the applicable Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER),[1] Millstone Unit No. 2 needs to fulfill two requirements to submit a plant-specific application for changes to RCP flywheel inspection requirements. The two requirements are: (1) verify the reference temperature $RT_{NDT}$ for its RCP flywheels, and (2) for the four flywheels made of ASTM-A 516, Grade 70 justify the use of the $K_{Ic}$ v.s. ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code to derive their respective $K_{Ic}$ values. Dominion Nuclear Connecticut, Inc. (DNC) performed a detailed technical evaluation which concluded that Millstone Unit No. 2 meets the requirements prescribed in the NRC SER for the Combustion Engineering Owners Group Topical report SIR-94-080-A[2] and may submit a plant-specific application for changes to RCP flywheel inspection requirements.

Discussion

(1) Verify the reference temperature $RT_{NDT}$ for RCP flywheels:

A review of material test information was performed to estimate $RT_{NDT}$ for each of the flywheel materials. Millstone Unit No. 2 has Reactor Coolant Pump (RCP) flywheels which were originally supplied with the plant which are produced to ASTM-A 516-69 Grade 65 and ASTM-A 300-68 specifications which is equivalent to SA-516 in Section II of the ASME Boiler and Pressure Vessel Code. In addition, it was noted that this material meets the minimum properties of SA-516 Grade 70 material with adequate margin and was named SA-516 Grade 70 material. This is vacuum improved plate material and was shown to be acceptable for this application as described in the FSAR. Replacement RCP motors (Serial Numbers 1044948 and 1046566) were purchased with flywheels fabricated to ASTM-A 508 grade 5.

In the case of the original flywheel material, A 516 Grade 65, the available strength and toughness information is summarized in Table 1.

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(1) B. W. Sheron (NRC) to D. C. Mims (Entergy), "Acceptance for Referencing of Topical Report SIR-94-080-A, Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements," dated May 21, 1997.

(2) Entergy Operations, Inc., letter from J. W. Yelverton (Entergy) to the Nuclear Regulatory Commission with enclosed report, SIR-94-080-A, "Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements," dated April 4, 1995.

Table 1
Material Information for Original RCP Flywheels
A 516 Grade 65

| Melt No. | Slab No. | Tensile Strength (psi) | 0.2% Offset Yield Strength (psi) | Charpy V-Notch @ 40°F[1] (ft-lb) | Drop Weight Test[2] |
|---|---|---|---|---|---|
| B3725 | 3 | 76700 | 49000 | 104, 109, 91 | acceptable |
| B3725 | 4 | 75600 | 50700 | 103, 95, 97 | acceptable |
| A8176 | 2 | 76700 | 51500 | 84, 73, 83 | acceptable |

[1]Longitudinal Charpy Orientation
[2]Drop weight tests were performed at 40°F and confirmed acceptable. An actual Nil-ductility Transition temperature based upon drop weights was not established.

The information obtained from the original material tests do not provide sufficient information to determine $RT_{NDT}$ in accordance with ASME Boiler and Pressure Vessel Code, Section III, NB-2331. However, using the guidance provided by the NRC for Class 1 pressure vessel materials, a conservative estimate can be established. Position 1.1(4) of MTEB 5-2 states that the $RT_{NDT}$ can be estimated as the test temperature when multiple Charpy tests were performed at a single test temperature to demonstrate 30 ft-lb for longitudinal oriented Charpy specimens provided at least 45 ft-lbs was obtained. Each of the material heats and individual slabs were tested at 40°F with Charpy values much greater than 45 ft-lbs and can therefore be conservatively estimated to have an $RT_{NDT}$ of 40°F. While MTEB 5-2 is for use with SA-508 Class II forgings and SA-533 Grade B Class 1 plate materials, use of this method should be conservative since the drop weight tests conservatively indicate a Nil-ductility Transition Temperature (NDTT) of 40°F and at the Nil-ductility Transition (NDT) test temperature the Charpy values are approximately 100 ft-lbs.

In the case of the replacement RCP flywheel (RCP motor serial number 1044948), the strength and toughness material tests are summarized in Table 2.

Table 2
Material Information for Replacement RCP Flywheel S/N 1044948
A 508 Class 5

| Heat and Ingot No. | Tensile Strength (psi) | 0.2% Offset Yield Strength (psi) | Charpy V-Notch @ 19°F[3] (ft-lb) | Lateral expansion (mils) | Shear (%) | Drop Weight Test[4] |
|---|---|---|---|---|---|---|
| 515 100/0101 | 112694 | 99786 | 142, 137, 133 | 89, 86, 83 | 100 | acceptable |
| 515 100/0101 | 112259 | 96015 | 140, 130, 122 | 90, 82, 80 | 100 | acceptable |

[3]Transverse Charpy Orientation
[4]Drop weight tests were performed at -31°F which were recorded as no break

In the case of the A 508 Class 5 forging, a conservative estimation of $RT_{NDT}$ can also be made from the available data using the criteria provided by ASME Boiler and Pressure Vessel Code Section III, NB-2331. A drop weight test temperature ($T_{NDT}$) can be established as -31°F for the forging. The actual value may be lower. The Charpy tests performed were at a temperature of 19°F ($T_{NDT}$ + 50°F) which satisfies the test temperature of "not greater than $T_{NDT}$ + 60°F" and exhibited greater than 50 ft-lb and 35 mils lateral expansion. Therefore, $RT_{NDT}$ can be conservatively established as $T_{NDT}$, or -31°F.

In the case of the second replacement RCP flywheel (RCP motor serial number 1046566), the strength and toughness material tests are summarized in Table 3.

Table 3
Material Information for Replacement RCP Flywheel S/N 1046566
A 508 Class 5

| Heat No. | Tensile Strength (psi) | 0.2% Offset Yield Strength (psi) | Charpy V-Notch[5,6] (ft-lb) | Lateral expansion (mils) | Drop Weight Test[7] |
|---|---|---|---|---|---|
| E320647 | 110084 | 93694 | 130, 126, 93 | 81, 77, 53 | -49 |
| E320647 | 111244 | 94855 | 134, 124, 131 | 70, 74, 84 | -49 |

[5]Transverse Charpy Orientation
[6]Test temperature was 20°F.
[7]Drop weight tests were performed and NDTT was reported as ≤-49°F. No breaks were reported.

In this instance, $RT_{NDT}$ can be established from the available data using the criteria provided by ASME Boiler and Pressure Vessel Code Section III, NB-2331 for the A 508 Class 5 forging. A drop weight test temperature ($T_{NDT}$) is established as -49°F for the forging. The actual value may be lower. The Charpy tests performed were at a temperature of 20°F ($T_{NDT}$ + 70°F). While all Charpy impact energies were much greater than 50 ft-lb and exhibited greater than 50 mils lateral expansion, the requirement which specifies the test temperature of "not greater than $T_{NDT}$ + 60°F" was not demonstrated by the existing data. However, $RT_{NDT}$ can be established as the higher of $T_{NDT}$ or $T_{CV}$ - 60°F. Consequently, $T_{CV}$ - 60°F = 20°F - 60°F = -40°F. Therefore, the $RT_{NDT}$ of the RCP flywheel (RCP motor serial number 1046566) is established as -40°F.

(2) For the four flywheels made of ASTM-A 516, Grade 70 justify the use of the $K_{Ic}$ v.s. ($T-RT_{NDT}$) curve in Appendix A of Section XI of the ASME Code to derive their respective $K_{Ic}$, values:

To demonstrate that the ASME Boiler and Pressure Vessel Code, Section XI, Appendix A $K_{Ic}$ curve is conservative and applicable to the A 516 Grade 65/70 plate material, conversion of the Charpy V-notch impact data was performed. Conversion of the Charpy V-notch data will provide a single temperature data point to determine if the curve provided by the text appears conservative or representative. To convert the data, two correlations were used. The first, developed by Corten and Sailors and the second, developed by Roberts and Newton, both of which ignore strain rate effects and would be expected to be conservative. Both equations are used to provide relative magnitude of the available fracture toughness although the correlation developed by Roberts and Newton is considered a lower bound.

The correlation developed by Corten and Sailors to estimate $K_{Ic}$ from Charpy V-notch impact values is given by the following expression:

$K_{Ic} = 15.5(CVN)^{0.5}$ where
$K_{Ic}$ = critical stress intensity factor, ksi$\sqrt{in}$
CVN = Charpy V-notch impact energy, ft-lb

The expression derived by Roberts and Newton can be expressed as:

$K_{Ic} = 9.35(CVN)^{0.63}$ where
$K_{Ic}$ = critical stress intensity factor, ksi$\sqrt{in}$
CVN = Charpy V-notch impact energy, ft-lb

To estimate the critical stress intensity from the Charpy V-notch impact data provided in Table 1, the values will be reduced to sixty five percent to estimate the transverse Charpy energy from the longitudinal values. Then using the above correlations, estimates of the critical stress intensity factor, $K_{Ic}$, can be calculated and are provided in Table 4.

Table 4
Critical Stress Intensity Factors, $K_{Ic}$, at 40°F
RCP Flywheel Material A 516 Grade 65

| Melt No. | Slab No. | Charpy V-Notch @ 40°F (ft-lb) | $K_{Ic}$, Corten and Sailors Critical Stress Intensity Factors From CVN, (ksi$\sqrt{in}$) | $K_{Ic}$, Roberts and Newton Critical Stress Intensity Factors From CVN, (ksi$\sqrt{in}$) |
|---|---|---|---|---|
| B3725 | 3 | 67.6, 70.9, 59.1 | 127.4, 130.5, 119.2 | 132.9, 137.0, 122.2 |
| B3725 | 4 | 67.0, 61.8, 63.1 | 126.9, 121.9, 123.1 | 132.2, 125.6, 127.3 |
| A8176 | 2 | 54.6, 47.5, 54.0 | 114.5, 106.8, 113.9 | 116.2, 106.4, 115.4 |

These critical stress intensity factors compare favorably with the value calculated from the $K_{Ic}$ curve to Section XI of the ASME Boiler and Pressure Vessel Code. ASME Section XI Appendix A, Figure A-4200-1 provides lower bound fracture toughness values for SA-533 Grade B Class 1, SA-508 Class 2 and Class 3 steels. Given an $RT_{NDT}$ of 40°F and a operating temperature of 40°F ($T-RT_{NDT}=0$), the value of $K_{Ic}$ is approximately 54 ksi$\sqrt{in}$. If a normal operating temperature of 100°F is used ($T-RT_{NDT}=60°F$), a value of 102 ksi$\sqrt{in}$ is obtained for $K_{Ic}$.

The significance of this comparison is two fold. First, the critical stress intensity values of the actual material at the Charpy test temperature demonstrate that the toughness is greater than the lower bound curve provided by ASME Section XI.

Secondly, the critical value provided by ASME Section XI of 102 ksi$\sqrt{in}$ at the normal operating temperature of 100°F exceeds the value used in the fracture mechanics analysis of 90 ksi$\sqrt{in}$ by 13.3 percent further increasing the margin of safety.

Conclusion

The $RT_{NDT}$ for each of the RCP flywheels has been established based upon the information available. The maximum $RT_{NDT}$ was established as +40°F and is acceptable.

A review of the available Charpy test information associated with flywheel fabricated of ASTM-A-516 Grade 70 was performed. This information was used to conservatively estimate $K_{Ic}$ based on analytical correlation. These estimates demonstrated that the fracture toughness of each flywheel is bounded by the lower bound values of $K_{Ic}$ curve.