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The Northeast Utilities System

SEP 9 1998

Docket No. 50-336
B17381

Re: 10CFR50.90

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

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Compliance Issues Number 4

Introduction

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2. NNECO is proposing to change Technical Specification Definitions 1.24, "Core Operating Limits Report," 1.27, "Engineering Safety Feature Response Time," and 1.31, "Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)," Technical Specifications 3.0.2, 4.0.5, 3.2.3, "Power Distribution Limits - Total Unrodded Integrated Radial Peaking Factor - F_r^T," 3.3.2.1, "Instrumentation - Engineered Safety Feature Actuation System Instrumentation," 3.4.1.1, "Reactor Coolant System - Coolant Loops and Coolant Circulation Startup and Power Operation," and 3.4.11, "Reactor Coolant System - Reactor Coolant System Vents." Technical Specification 3.0.6 will be added. Information will be added to the Bases of the associated Technical Specifications to address the proposed changes.

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.

The proposed change to Technical Specification Section B 3/4.3.1 and 3/4.3.2 is on the same page (B 3/4 3-1) which has been proposed to be changed in a separate letter

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dated April 6, 1998,[1] which addressed Reactor Protective System and Engineered Safety Feature Actuation System issues. The proposed changes contained in this letter do not assume approval of any of the previously submitted changes.

Environmental Considerations

NNECO has reviewed the proposed License Amendment Request against the criteria of 10CFR51.22 for environmental considerations. The proposed changes modify a surveillance requirement for the Reactor Coolant System vent valves. These changes do not increase the type and amounts of effluents that may be released off site. In addition, this amendment request will not significantly increase individual or cumulative occupational radiation exposures. Therefore, NNECO has determined the proposed changes will not have a significant effect on the quality of the human environment.

Conclusions

The proposed changes were evaluated utilizing the criteria of 10CFR50.59 and were determined not to involve an unreviewed safety question. In addition, we have concluded the proposed changes are safe.

The proposed changes do not involve a significant impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10CFR50.92 (see the Significant Hazards Consideration provided in Attachment 2).

Plant Operations Review Committee and Nuclear Safety Assessment Board

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

Schedule

We request issuance at your earliest convenience, with the amendment to be implemented within 60 days of issuance.

State Notification

In accordance with 10CFR50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

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(1) M. L. Bowling, Jr. letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2 Proposed Revision to Technical Specifications Reactor Protective and Engineered Safety Feature Actuation System Instrumentation," dated May 14, 1998.

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

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Martin L. Bowling, Jr.
Recovery Officer - Technical Services

BY: John P. McElwain
John P. McElwain
Recovery Officer - Millstone Unit No. 2

Sworn to and subscribed before me

this 9 day of September, 1998

Loretta F. Goodson
Notary Public

My Commission expires LORETTA F. GOODSON
NOTARY PUBLIC
Commission Expires November 30, 2001

Attachments (5)

cc: H. J. Miller, Region I Administrator
D. G. McDonald, Jr., NRC Senior Project Manager, Millstone Unit No. 2
D. P. Beaulieu, Senior Resident Inspector, Millstone Unit No. 2
W. M. Dean, Director, Millstone Project Directorate
W. D. Lanning, Director, Millstone Inspections
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Docket No. 50-336
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Attachment 1

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Compliance Issues Number 4
Discussion of Proposed Changes

September 1998

Proposed Revision to Technical Specifications
Compliance Issues Number 4
Discussion of Proposed Changes

Introduction

Northeast Nuclear Energy Company (NNECO) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2. NNECO is proposing to change Technical Specification Definitions 1.24, "Core Operating Limits Report," 1.27, "Engineering Safety Feature Response Time," and 1.31, "Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)," Technical Specifications 3.0.2, 4.0.5, 3.2.3, "Power Distribution Limits - Total Unrodded Integrated Radial Peaking Factor - F_r^T," 3.3.2.1, "Instrumentation - Engineered Safety Feature Actuation System Instrumentation," 3.4.1.1, "Reactor Coolant System - Coolant Loops and Coolant Circulation Startup and Power Operation," and 3.4.11, "Reactor Coolant System - Reactor Coolant System Vents." Technical Specification 3.0.6 will be added. Information will be added to the Bases of the associated Technical Specifications to address the proposed changes.

Description of Proposed Changes

Technical Specification Definitions

1.24 Core Operating Limits Report

The reference to Specification 6.9.1.7, "Monthly Operating Report," contained in this definition is not correct. Specification 6.9.1.8, "Core Operating Limits Report," should be referenced. The proposed change will correct this error.

1.27 Engineering Safety Feature Response Time

The use of "Engineering" in the title is not correct. This should be "Engineered," which is the correct terminology for this system as indicated in the first line of this definition and in the title of Technical Specification 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation." The proposed change will correct this error.

1.31 Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)

The word "radionuclines" is not spelled correctly. The correct spelling is "radionuclides." The proposed change will correct this error.

The reference to Specification 6.16, "Radioactive Waste Treatment," contained in this definition is not correct. Specification 6.15, "Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM)," should be referenced. The proposed change will correct this error.

Technical Specifications 3.0.2 and 3.0.6

The proposed change will add Technical Specification 3.0.6. This new specification will state that it is acceptable to return inoperable equipment to service, under administrative control, but only to demonstrate operability of that equipment, or the operability of other equipment. Since this is an exception to Technical Specification 3.0.2, a reference to Technical Specification 3.0.6 will be added to Technical Specification 3.0.2. This change is consistent with NUREG-1432.[(1)]

The Bases will be expanded to discuss the new proposed specification.

Technical Specification 4.0.5 and Bases 3/4.4.10

The proposed changes will revise Technical Specification 4.0.5.a and Bases 3/4.4.10, "Structural Integrity," by removing the phrase "(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i)." The changes relate to inservice inspection (ISI) and inservice testing (IST) requirements which are specified in 10CFR50.55a, "Codes and Standards." The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code is incorporated by reference in the regulation as the requirements for ISI and IST.

NUREG-1482[(2)] addresses the situation in which the Technical Specifications are in conflict with the regulations of 10CFR50.55a. As discussed in NUREG-1482, the NRC has recognized that situations could arise which would put the licensee in a condition that is not in strict compliance with Technical Specification 4.0.5 requirements to comply with ASME Section XI "except where specific written relief has been granted." According to NUREG-1482, if Technical Specification 4.0.5 was interpreted literally, in the case of the IST Program, it could require the licensee to address these situations by shutting the plant down to perform testing. To correct this, NUREG-1482 recommends that Technical Specifications be revised. The proposed changes are consistent with the recommendations of NUREG-1482.

(1) NUREG - 1432, Standard Technical Specifications Combustion Engineering Plants, Revision 1, April 1995.

(2) NUREG - 1482, Guidelines for Inservice Testing at Nuclear Power Plants, April 1995.

Technical Specification 3.2.3

The proposed change will change the mode of applicability for Technical Specification 3.2.3, "Total Unrodded Integrated Radial Peaking Factor - F_r^T," from "MODE 1*" to "MODE 1 with THERMAL POWER > 20% RTP*." Data from the incore detectors are used for determining the measured radial peaking factors. However, the accuracy of the neutron flux information from the incore detectors is not reliable below 20% power. The proposed change acknowledges this limitation of the incore detectors by changing the applicability of this specification to power levels where the data from the incore detectors is reliable.

The Bases for this specification, as well as Technical Specification 3.2.1, will be expanded to discuss the accuracy of the incore detectors at low power levels. In addition the power level listed in the Bases for Technical Specification 3.2.3 will be changed from "75%" to "70%" to agree with the requirements of Surveillance Requirement (SR) 4.2.3.2.a.

Technical Specification 3.3.2.1

The channel functional test requirement for the automatic actuation logic associated with Engineered Safety Feature (ESF) actuations for safety injection, containment spray, containment isolation, main steam line isolation, enclosure building filtration, and containment sump recirculation will be modified by adding an exception to Technical Specification 4.0.4. The automatic actuation logic for these functions is normally tested by use of the Automatic Testing Insertor (ATI) circuit. However, the ATI will not function properly when the features checked by the ATI are blocked or bypassed. During plant startup, the low pressurizer pressure safety injection and the low steam line pressure main steam line isolation actuations are blocked until pressurizer pressure and steam generator pressure have been raised sufficiently to automatically remove the blocks. Since this does not normally occur until after Mode 3 is entered, the channel functional test of the automatic actuation logic for these features cannot be performed by use of the ATI circuit. This has resulted in a violation of Technical Specification 4.0.4, as reported by Licensee Event Report 98-008-00.[3]

The proposed change will allow entry into Mode 3 without performing the channel functional test of the automatic actuation logic. The channel functional test of the automatic actuation logic will be performed after the blocks on low pressurizer pressure and low steam generator have been removed. It is expected the tests will be performed within 24 hours of establishing the

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(3) J. A. Price letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2, Licensee Event Report 98-008-00, Technical Specification Violations, dated May 26, 1998.

necessary plant conditions. The tests must be performed prior to entering Mode 2.

The Bases will be expanded to discuss the proposed change.

Technical Specification 3.4.1.1

The proposed change will replace SR 4.4.1.1 with a verification requirement that is more consistent with the Limiting Condition for Operation (LCO). The current verification requirement indirectly verifies that two Reactor Coolant System (RCS) loops and four Reactor Coolant Pumps (RCPs) are in operation. With the Flow Dependent Setpoint Selector Switch in the 4 pump position, a reactor trip will occur if less than 4 RCPs are operating. Therefore, the plant will not remain in Mode 1 or 2 unless all 4 RCPs are operating.

The proposed wording change to SR 4.4.1.1 is consistent with the current LCO, and is consistent with NUREG-0212[4] and NUREG-1432.

Technical Specification 3.4.11

SR 4.4.11.3 will be modified by removing the words "during venting." This change is necessary because the current wording requires that flow through the entire reactor vessel head and pressurizer vent paths be verified in Modes 5 and 6. The vent paths discharge through a sparger directly into the containment structure. If flow is initiated and released through the sparger into containment, this will result in possible contamination of the surrounding area. In addition, to obtain water flow through the pressurizer vent paths would require the establishment of solid water conditions in the RCS. This would significantly increase the potential for a cold overpressure event.

An alternate approach to verifying flow is to use a series of overlapping tests to verify flow through all sections of the vent system, such that when completed, flow will be verified through all parts of the vent system. An alternate water source may be used, when necessary, to provide the fluid for flow verification.

The failure to perform the surveillance test as written, a violation of Technical Specifications, was report by Licensee Event Report 97-007-00.[5]

The Bases will be expanded to discuss flow testing of the vent system.

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[4] NUREG - 0212, Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors, Revision 2, Fall 1980.

[5] J. A. Price letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2, Licensee Event Report 97-007-00, Inadequate Surveillance Procedure for Verifying Operability of Reactor Coolant System Vents, dated April 4, 1997.

Safety Summary

Technical Specification Definitions

The minor editorial and non-technical changes to correct reference, spelling and terminology errors contained in the definitions will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes will have no adverse effect on plant operation. Therefore, there will be no adverse impact on public health and safety.

Technical Specification 3.0.6

The addition of TS 3.0.6 will allow inoperable equipment to be placed in a condition different from that required by the Action Statement to demonstrate the operability of that equipment or other equipment. This provision is provided only to perform surveillance requirements to prove operability, and not to provide time to perform any other preventive or corrective maintenance. The testing will be performed consistent with the current Technical Specification Action Statement and will be limited to the time necessary to perform the surveillance requirement. The proposed changes will have no adverse effect on plant operations. Therefore, there will be no adverse impact on public health and safety. The proposed changes are consistent with NUREG-1432.

Technical Specification 4.0.5

The proposed revision to Technical Specification 4.0.5 will allow the licensee to follow the provisions of 10CFR50.55a for a relief request upon finding an ASME Code requirement impractical because of limitations in the design (including prohibitive dose rates), construction, or system configuration. In accordance with 10CFR50.55a(f)(5)(iv) and 10CFR50.55a(g)(5)(iv), a licensee has up to a full year after the start of a new interval to inform the NRC of those new code requirements which cannot be met and to request relief. This will eliminate inconsistencies between the Technical Specifications and the regulations. The proposed changes will have no adverse effect on plant operations. Therefore, there will be no adverse impact on public health and safety. The proposed changes are consistent with NUREG-1482.

Technical Specification 3.2.3

The proposed change will change the mode of applicability for Technical Specification 3.2.3 from Mode 1 to Mode 1 with thermal power > 20%. Data from the incore detectors are used for determining the measured radial peaking factors to verify compliance with Technical Specification 3.2.3. However the accuracy of the neutron flux information from the incore detectors is not reliable below 20% power. The proposed change acknowledges this limitation of the incore detectors by changing the applicability of this specification to power levels where the data from the incore detectors is reliable. This will have no adverse effect on plant operations since the current Technical Specification surveillance requirements do not require the verification of this limit until prior to operation above 70% following each fuel loading, prior to 31 days accumulated operation in Mode 1, or if the azimuthal power tilt limit is exceeded (Technical Specification 3.2.4 which is applicable in Mode 1 above 50% power). Therefore, there will be no adverse impact on public health and safety.

Technical Specification 3.3.2.1

The proposed change will add an exception to Technical Specification 4.0.4 that will allow the channel functional test of the automatic actuation logic associated with ESF actuations for safety injection, containment spray, containment isolation, main steam line isolation, enclosure building filtration, and containment sump recirculation to be delayed during plant startup until the actuation blocks are removed. This will allow entry into Mode 3 where plant conditions (sufficient pressurizer and steam generator pressure) can be established that will automatically remove the blocks of these ESF actuations. The channel functional tests of the automatic actuation logic, using the ATI circuit, will then be performed. In addition, the channel functional tests of the automatic actuation logic must be performed prior to entering Mode 2. Even though operability of the automatic actuation logic for the affected ESF actuations cannot be verified prior to entering Mode 3, this equipment is still expected to be operable. The Engineered Safety Feature Actuation System (ESFAS) will continue to function as before. Therefore, there will be no adverse impact on public health and safety.

Technical Specification 3.4.1.1

The proposed change will replace SR 4.4.1.1 with a verification requirement that is more consistent with the Limiting Condition for Operation (LCO). This will not change the requirement that both RCS loops be operable and operating in Modes 1 and 2. The RCS will continue to function as before. Therefore, there will be no adverse impact on public health and safety. The proposed change is consistent with NUREG-0212 and NUREG-1432.

Technical Specification 3.4.11

The proposed change to modify the wording of SR 4.4.11.3 will not affect the operability requirements of the RCS Vent System. This change will provide operational flexibility to use a series of overlapping tests to verify flow through sections of the vent system, such that when completed, flow will be verified through all parts of the vent system. This will minimize potential contamination of the area surrounding the sparger and will eliminate the need to establish solid water conditions in the RCS.

The proposed surveillance requirement will still verify the ability of the vent valves to operate. This will provide reasonable assurance of system operability and availability if needed to mitigate the consequences of design basis accidents. Therefore, there is no adverse impact on public health and safety.

The proposed changes have no adverse effect on how any of the associated systems or components function to prevent or mitigate the consequences of design basis accidents. Also, the proposed changes have no adverse effect on any design basis accident previously evaluated. Therefore, there is no adverse impact on public health and safety.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Compliance Issues Number 4
Significant Hazards Consideration

September 1998

Proposed Revision to Technical Specifications
Compliance Issues Number 4
Significant Hazards Consideration

Significant Hazards Consideration

In accordance with 10CFR50.92, NNECO 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 10CFR50.92(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.

    Technical Specification Definitions

    The minor editorial and non-technical changes to correct reference, spelling and terminology errors contained in the definitions will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes will have no adverse effect on plant operation. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

    Technical Specification 3.0.6

    The new Technical Specification, 3.0.6, will provide guidance on returning inoperable equipment to service, under administrative control, to demonstrate operability of that equipment, or the operability of other equipment. Various Technical Specification Actions require inoperable equipment to be removed from service, such as maintaining a containment isolation valve closed or tripping/bypassing a failed instrument channel. An exception to these required actions is necessary to allow the performance of testing to demonstrate the operability of the equipment being returned to service. Specifically, this Technical Specification addresses the situation where the inoperable equipment has been repaired, tested to the extent possible, and believed to be capable of performing its function. At this point, a presumption of the operability of the equipment is reasonable, and is supported by experience. Therefore, it is acceptable to place the equipment in service for testing under administrative control. Administrative controls will be used to ensure the time the equipment is returned to service is consistent with the Action Statements and is limited to the time necessary to perform the surveillance requirements.

    This specification will also allow the inoperable equipment to be placed in a condition different from that required by the action statement to demonstrate the operability of other equipment. An example would be during the performance of an operability test on one reactor protection channel while another channel associated with the same function is inoperable. In this situation only one of the channels could be in the tripped condition, otherwise a reactor trip would be initiated. This is already permitted for reactor protection channels by Technical Specifications 3.3.1.1, "Instrumentation - Reactor Protective Instrumentation," Action 2, and for engineered safety features channels by 3.3.2.1, "Instrumentation - Engineered Safety Feature Actuation System Instrumentation," Action 2.

This provision is provided only to perform surveillance requirements to prove operability, and not to provide time to perform any other preventive or corrective maintenance. The testing will be performed consistent with the current Technical Specification Action Statement and will be limited to the time necessary to perform the surveillance requirement. The proposed changes will have no adverse effect on plant operations. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 4.0.5

The proposed changes will revise Technical Specification 4.0.5.a and Bases 3/4.4.10, "Structural Integrity," by removing the phrase "(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i)." The changes to Technical Specifications clarify that all applicable requirements in 10CFR50.55a apply. The changes relate to inservice inspection (ISI) and inservice testing (IST) requirements which are specified in 10CFR50.55a, "Codes and Standards." The ISI and IST requirements are given in 10CFR50.55a, which the licensee documents via its 10 year interval program requirements. Upon finding a Code requirement impractical because of limitations in the design (including prohibitive dose rates), construction, or system configurations, NNECO would be required to prepare the determination describing the impractical condition(s) and the applicable code requirements that cannot be met in accordance with 10CFR50.55a, paragraphs (f)(5)(iii) and (iv), and (g)(5)(iii) and (iv) if within the first 12 months of a new interval. For example, 10CFR50.55a(f)(5)(iv), and (g)(5)(iv) allow a licensee up to a full year after the beginning of an updated interval to inform the NRC of the new Code requirements which cannot be met and to request relief. If an impracticality is identified after the first 12 months, the guidance contained in NUREG-1482 will be followed. This will eliminate inconsistencies between the Technical Specifications and the regulations. There will be no adverse effect on plant operations. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.2.3

The proposed change will change the mode of applicability for Technical Specification 3.2.3 from Mode 1 to Mode 1 with thermal power > 20%. Data from the incore detectors are used for determining the measured radial peaking factors to verify compliance with Technical Specification 3.2.3. However the accuracy of the neutron flux information from the incore detectors is not reliable below 20% power. The proposed change acknowledges this limitation of the incore detectors by changing the applicability of this specification to power levels where the data from the incore detectors is reliable. This will have no adverse effect on plant operations since the current Technical Specification surveillance requirements do not require the verification of this limit until prior to operation above 70% following each fuel loading, prior to 31 days accumulated operation in Mode 1, or if the azimuthal power tilt limit is exceeded (Technical Specification 3.2.4 which is applicable in Mode 1 above 50% power). Therefore, the proposed change has no impact on the initial conditions, with respect to power distribution, assumed in the accident analysis. Thus, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.3.2.1

The proposed change will add an exception to Technical Specification 4.0.4 that will allow the channel functional test of the automatic actuation logic associated with ESF actuations for safety injection, containment spray, containment isolation, main steam line isolation, enclosure building filtration, and containment sump recirculation to be delayed during plant startup until the actuation blocks are removed. This will allow entry into Mode 3 where plant conditions (sufficient pressurizer and steam generator pressure) can be established that will automatically remove the blocks of these ESF actuations. The channel functional tests of the automatic actuation logic, using the ATI circuit, will then be performed. In addition, the channel functional tests of the automatic actuation logic must be performed prior to entering Mode 2.

The exception to Technical Specification 4.0.4 allows a mode change with equipment that is inoperable only because conditions can not be established to perform the SR until after the mode is entered. All other equipment operability requirements must be met. Even though operability of the automatic actuation logic for the affected ESF actuations cannot be verified prior to entering Mode 3, this equipment is still expected to be operable. The ESFAS will continue to function as before. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.1.1

The Flow Dependent Setpoint Selector Switch was installed to allow power operation with less than four reactor coolant pumps (RCPs) in operation by changing the reactor trip setpoints for the variable high power, Reactor Coolant System (RCS) low flow, and thermal margin low pressure (TM/LP) reactor trips. Millstone Unit No. 2 is not currently licensed to operate with less than four RCPs in operation. Therefore, this switch should be maintained in the four pump position.

The use of the switch position to ensure compliance with Technical Specification 3.4.1.1 provides an indirect verification of LCO compliance since the loss of an RCP will result in a reactor trip when in the four pump position. The proposed change will replace the method used for LCO verification with one that is more consistent with the LCO. Verification of switch position is performed as a prerequisite prior to reactor startup (entering Mode 2). It is not necessary to verify the switch position every 12 hours as currently required. The position of this switch is important to the operability of the associated Reactor Protection System (RPS) trips (variable high power, RCS low flow, and TM/LP). The operability of these RPS trips and associated setpoints is already covered by Technical Specifications 2.2.1, "Reactor Trip Setpoints," and 3.3.1.1, "Reactor Protective Instrumentation."

It is not necessary to verify the position of this switch fifteen minutes prior to reactor criticality since the switch position is verified prior to a reactor startup, and is not expected to be changed during power operation. If surveillance testing or maintenance activities are to be performed which may require the switch to be in other than the four pump position, the affected RPS channels will already have been removed from service (declared inoperable and placed in the tripped or bypassed condition) prior to commencing the activities. In addition, a light ("PUMP SETPOINT ERROR") on each of the RPS Calibration and Indication Panels will illuminate if the switch is not in the four pump position.

It is also not necessary to verify compliance with the requirements of Technical Specification 3.4.1.1 within fifteen minutes prior to reactor criticality since this condition is verified prior to a reactor startup, and the RPS will initiate a reactor trip if less than four RCPs are in operation.

The proposed change will replace SR 4.4.1.1, verification of the Flow Dependent Setpoint Selector Switch position, with a verification check of the required RCS loops. This verification is more consistent with the Limiting Condition for Operation (LCO). This will not change the requirement that both RCS loops be operable and operating in Modes 1 and 2. The Technical Specification will continue to assure that the initial condition, with respect to RCS loops in service, in the accident analysis is applicable. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.11

The proposed change to modify the wording of SR 4.4.11.3 will not affect the operability requirements of the RCS Vent System. This change will provide operational flexibility to use a series of overlapping tests to verify flow through sections of the vent system, such that when completed, flow will be verified through all parts of the vent system. This will minimize potential contamination of the area surrounding the sparger and will eliminate the need to establish solid water conditions in the RCS.

The proposed surveillance requirement will still verify the ability of the vent valves to operate. This will provide reasonable assurance of system operability and availability if needed to mitigate the consequences of design basis accidents. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on any of the design basis accidents previously evaluated or on any equipment important to safety. Therefore, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the 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. 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 changes will correct reference, spelling, and terminology errors in various Technical Specification Definitions; add a new Technical Specification, 3.0.6; modify Technical Specification 4.0.5 to remove an inconsistency between the Technical Specification and the regulations; change the applicability of Technical Specification 3.2.3; add an exception to Technical Specification 4.0.4 to Technical Specification 3.3.2.1; modify the wording of a surveillance requirement associated with RCS Technical Specification 3.4.1.1; and modify the wording of a surveillance requirement associated with the RCS Vent System, Technical Specification 3.4.11 to provide operational flexibility in the performance of the test. These changes will have no adverse effect on equipment important to safety. The equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction of the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

The NRC has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51 FR 7751) of amendments that are considered not likely to involve an SHC. The minor editorial and non-technical changes proposed herein to correct reference, spelling, and terminology errors are enveloped by example (i), a purely administrative change to Technical Specifications. All of the other changes proposed herein are not enveloped by any specific example.

As described above, this License Amendment Request does not impact the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO has concluded that the proposed changes do not involve an SHC.

Docket No. 50-336
B17381

Attachment 3

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Compliance Issues Number 4
Marked Up Pages

September 1998