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The Northeast Utilities System

SEP 28 1998

Docket No. 50-336
B17413

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
Control Room Ventilation System

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 Specifications 3.3.2.1, "Instrumentation – Engineered Safety Features Actuation System;" 3.4.6.2, "Reactor Coolant System – Reactor Coolant System Leakage;" 3.4.8, "Reactor Coolant System – Specific Activity;" 3.6.2.1, "Containment Systems - Depressurization and Cooling Systems Containment Spray and Cooling Systems;" 3.6.5.1, "Containment Systems – Secondary Containment Enclosure Building Filtration System;" 3.7.6.1, "Plant Systems – Control Room Emergency Ventilation System;" and 3.9.15, "Refueling Operations – Storage Pool Area Ventilation System – Fuel Storage." Information will be added to the Bases of the associated Technical Specifications to address the proposed changes.

NNECO also proposes to amend Operating License DPR-65 by incorporating the attached change to the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). The change to the FSAR is associated with the revised main steam line break analyses, new determination of the radiological consequences of a main steam line break, and a revised determination of the radiological consequences of the design basis loss of coolant accidents (LOCAs).

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Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the marked-up version of the appropriate pages of the current Technical Specifications. Attachment 4 provides the retyped pages of the Technical Specifications. Attachment 5 provides the change to the Millstone Unit No. 2 FSAR.

The proposed changes to the main steam line break analysis, as described in the FSAR, are based on the revised Siemens Power Corporation steam line break methodology. The report describing the revised methodology was submitted by Siemens Power Corporation to the NRC for approval in a letter dated June 30, 1998.(1) The revised methodology was used to perform the Millstone Unit No. 2 plant specific analysis for post-scram main steam line break. This plant specific analysis(2) was submitted by NNECO in a letter dated August 12, 1998,(3) which proposed to change the list of documents in Technical Specifications which describe the analytical methods used to determine the core operating limits. The proposed changes contained in this letter assume approval of the previously submitted revised Siemens Power Corporation steam line break methodology, and the changes to the list of documents in the Millstone Unit No. 2 Technical Specifications which describe the analytical methods used to determine the core operating limits.

The determination of the radiological consequences to the Millstone Unit No. 2 Control Room Operators from a Millstone Unit No. 3 LOCA is based on the expected release to the environment taking credit for proper operation of the Millstone Unit No. 3 Supplementary Leak Collection Release System (SLCRS). A License Amendment Request,(4) concerning the SLCRS is presently under review by the NRC. The resolution of the Millstone Unit No. 3 License Amendment Request may affect the Millstone Unit No. 2 radiological consequences. NNECO will advise the NRC if any changes to this License Amendment Request are necessary based on the Millstone Unit No. 3 submittal.

Environmental Considerations

NNECO has reviewed the proposed License Amendment Request against the criteria of 10CFR51.22 for environmental considerations. The proposed changes will correct spelling and terminology errors, reduce the maximum allowable primary to secondary leakage, add a new surveillance requirement, modify surveillance requirements for RCS specific activity, reduce the allowed outage time for a containment spray train, reduce the allowed pressure drop across the control room and enclosure building HEPA filters, and increase the control room maximum allowed in-leakage. The proposed changes also include a revised main steam line break analyses, new

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(1) J. F. Mallay, Siemens Power Corporation, letter to the NRC, "Request for review of ANF-84-093(P) Revision 1, 'Steamline Break Methodology for PWRs'," dated June 30, 1998.

(2) EMF-98-036, Revision 1, "Post-Scram Main Steam Line Break Analysis for Millstone Unit 2," July 1998, Siemens Power Corporation.

(3) M. L. Bowling, Jr. letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2 Proposed Revision to Technical Specifications Updating List of Documents describing the Analytical Methods Specified in Technical Specification 6.9.1b," dated August 12, 1998.

(4) M. H. Brothers letter to the NRC, "Millstone Nuclear Power Station, Unit No. 3 Proposed License Amendment Request SLCRS Bypass Leakage (PLAR 3-98-5)," dated June 6, 1998.

determination of the radiological consequences of a main steam line break, and a revised determination of the radiological consequences of the design basis LOCAs. The results meet the guidance contained in SRP 15.1.5,[5] SRP 15.6.5,[6] and the limits of 10CFR100 and 10CFR50, Appendix A, General Design Criteria (GDC) 19. 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 following items included in the proposed changes to the Technical Specifications and the FSAR involve unreviewed safety questions.

1. The increase in allowed control room in-leakage, from 100 standard cubic feet per minute (SCFM) to 130 SCFM, when the Control Room Emergency Ventilation System is operating in the recirculation/filtration mode.

2. The limited fuel failure that is predicted in the revised steam line break analysis (no fuel failure is predicted in the current analysis of record).

3. The credit taken for the Reactor Coolant System low flow reactor trip in a harsh containment environment in the steam line break inside containment analysis (low flow reactor trip is not credited in the current analysis of record).

4. The credit taken for removing radioactive iodines by the Containment Spray System from the post-Loss of Coolant Accident (LOCA) atmosphere (iodine scrubbing is not credited in the current analysis of record).

5. The addition of backleakage from the containment sump to the Refueling Water Storage Tank into the off-site and control room LOCA dose consequence analyses.

NNECO has concluded the proposed changes are safe. In addition, the proposed

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[5] Standard Review Plan (SRP) 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Rev.2 - July 1981.

[6] Standard Review Plan (SRP) 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within The Reactor Coolant Pressure Boundary," Rev.2 - July 1981.

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). Therefore, NNECO requests the NRC review and approve the proposed changes to the Millstone Unit No. 2 Technical Specifications and FSAR through an amendment to Operating License DPR-65, pursuant to 10CFR50.90.

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.

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,

NORTHEAST NUCLEAR ENERGY COMPANY

M L Bowling

M. L. Bowling, Jr.
Recovery Officer - Technical Services

Sworn to and subscribed before me

this 28 day of September, 1998

Loretta F. Goodson

Notary Public

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

Attachments (6)

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
Control Room Ventilation System
Discussion of Proposed Changes

September 1998

Proposed Revision to Technical Specifications
Control Room Ventilation System
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 Specifications 3.3.2.1, "Instrumentation – Engineered Safety Features Actuation System;" 3.4.6.2, "Reactor Coolant System – Reactor Coolant System Leakage;" 3.4.8, "Reactor Coolant System – Specific Activity;" 3.6.2.1, "Containment Systems - Depressurization and Cooling Systems Containment Spray and Cooling Systems;" 3.6.5.1, "Containment Systems – Secondary Containment Enclosure Building Filtration System;" 3.7.6.1, "Plant Systems – Control Room Emergency Ventilation System;" and 3.9.15, "Refueling Operations – Storage Pool Area Ventilation System – Fuel Storage." Information will be added to the Bases of the associated Technical Specifications to address the proposed changes.

NNECO, in a letter dated July 7, 1995,[1] proposed similar changes to the Millstone Unit No. 2 Technical Specifications. This request was withdrawn in a letter dated August 4, 1998.[2] However, during the review of the proposed changes the NRC requested additional information in separate letters dated November 6, 1997,[3] and November 25, 1997.[4] Since these proposed changes are very similar, the same questions may arise during the NRC review. Therefore, Attachment 6 contains NNECO's response to the questions.

Many of the proposed changes are the result of the revised main steam line break analyses and the revised determinations of the radiological consequences of the main steam line break and loss of coolant accident. A brief summary of the revised main steam line break analyses and the radiological consequences is presented to illustrate how the proposed changes are used in the revised analyses. Attachment 5 contains the associated change to the Millstone Unit No. 2 Final Safety Analysis Report (FSAR)

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(1) E. A. DeBarba letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2 Proposed Revision to Technical Specifications Control Room Ventilation System," dated July 7, 1995.

(2) M. L. Bowling, Jr. letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2 Retraction of Proposed Revision to Technical Specifications Regarding Control Room Ventilation System (TAC No. M92879)," dated August 4, 1998.

(3) D. G. McDonald Jr. letter to NNECO, "Request for Additional Information Relating to the Control Room Ventilation System, Millstone Nuclear Power Station, Unit No. 2 (TAC NO. M92879)," dated November 6, 1997.

(4) D. G. McDonald Jr. letter to NNECO, "Supplemental Request for Additional Information Relating to the Control Room Ventilation System, Millstone Nuclear Power Station, Unit No. 2 (TAC NO. M92879)," dated November 25, 1997.

which NNECO requests the NRC review and approve through an amendment to Operating License DPR-65.

Main Steam Line Break Analysis

The analysis of a main steam line break is divided into two categories. The first category, "Pre-Scram," evaluates plant response immediately after the break is initiated. The pre-scram evaluation does not extend long enough to include the effects of plant cooldown which may result in a return to criticality. The second category, "Post-Scram," evaluates long term plant response, including the effects of plant cooldown and the resultant return to criticality.

Pre-Scram

The pre-scram steam line break (SLB) event is initiated by a rupture in the main steam piping which results in an uncontrolled steam release from the secondary system.

Largest Linear Heat Generation Rate (FSAR Section 14.1.5.1.6.1)

The scenario that results in the highest power level and the largest linear heat generation rate (LHGR) is the hot full power (HFP) 3.50 ft^2 symmetric break outside containment, downstream of the check valve, with offsite power available for operation of the reactor coolant pumps.

The minimum departure from nucleate boiling ratio (MDNBR) value for this scenario was calculated to be 1.298, which is above the 95/95 XNB correlation limit of 1.17. Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR stand point.

The peak linear heat rate (LHR) for this LHR-limiting case (3.50 ft^2 break outside containment, downstream of the check valve) is calculated to be 19.7 kW/ft. Comparing this LHGR value with a centerline melt criteria of 21 kW/ft, centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

Minimum Departure from Nucleate Boiling Ratio (FSAR Section 14.1.5.1.6.2)

The scenario that results in the limiting MDNBR is the HFP 3.51 ft^2 asymmetric break inside containment, upstream of the check valve, with a loss of offsite power.

The MDNBR value for this scenario was calculated to be 0.88, which is below the 95/95 XNB correlation limit of 1.17. The number of failed assemblies is determined by comparing core power distribution to the assembly power where DNB occurs. This results in a predicted failure of 3.7% of the fuel rods in the core.

The peak LHR for this case is bounded by the 3.50 ft^2 break outside containment symmetric break (19.7 kW/ft). Therefore, the LHGR value is below the criteria of 21 kW/ft and no fuel failures are predicted to occur due to violation of the centerline melt criteria.

The pre-scram analysis now takes credit for the Reactor Coolant System (RCS) low flow reactor trip function for the main steam line breaks that assume a loss of offsite power. Since the break can occur inside containment it is necessary to qualify the RCS flow detectors and instrument loops for the harsh environment that is expected to develop in containment following a main steam line break. This work is in progress.

A summary of the Pre-Scram analysis results is contained in Table 1.

Table 1
MDNBR and Peak Reactor Power Level Summary
(Pre-Scram Steam Line Break)
(FSAR Table 14.1.5.1-6)

Location of Break	Type of Cooldown	Size of Break	MDNBR	Peak Reactor Power (% of rated)
Outside containment, downstream of check valves	Symmetric	2.40 ft^2	1.332	126.90%
		3.00 ft^2	1.310	130.01%
		3.50 ft^2	1.298	130.91%[5]
Outside containment, upstream of check valve	Asymmetric	1.20 ft^2	1.254	124.42%
		1.40 ft^2	1.244	126.06%
		1.60 ft^2	1.302	124.87%
		1.80 ft^2	1.334	124.92%
Inside containment, upstream of check valve	Asymmetric	0.40 ft^2	1.299	117.85%
		0.60 ft^2	1.258	121.53%
		0.80 ft^2	1.262	122.26%
		1.80 ft^2	1.318	125.51%
Inside containment, upstream of check valve with loss of offsite power	Asymmetric	3.51 ft^2	0.88[6]	106.86%

Post-Scram

The post-scram SLB event is initiated by a rupture in the main steam piping downstream of the integral steam generator flow restrictors and upstream of the main steamline isolation valves (MSIVs) which results in an uncontrolled steam release from the secondary system. The scenario that resulted in the highest post-scram power level and the largest LHGR is that initiated from HFP with the break occurring outside containment with offsite power available for operation of the reactor coolant pumps.

(5) The peak LHRs for all pre-scram breaks are bounded by the peak LHR for the 3.50 ft^2 break outside containment and downstream of a check valve.

(6) The MDNBRs for all pre-scram breaks are bounded by the MDNBR for the 3.51 ft^2 break inside containment and upstream of a check valve with a loss of offsite power.

Peak MDNBR and LHR (FSAR Section 14.1.5.2.6.1.5)

For the MDNBR portion of the calculation, the radial power distribution was modified to conservatively account for local rod power distribution affects within the hot assembly. This was done by raising the power of the hot assembly to bound the peak rod power. On the bases of these conservative assumptions, the MDNBR was calculated to be 2.28. This compares to the 95/95 DNBR limit of 1.135 for the modified Barnett correlation. Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR stand point.

The analysis of the peak LHGR also comes from the XTGPWR and XCOBRA-IIIC analysis. The peak LHGR is calculated from the ANF-RELAP total core power and the XTGPWR radial and axial peaking. The peak LHGR, 24.27 kW/ft, was calculated for the HFP outside containment break with offsite power available event. When compared to the centerline melt criteria of 21.0 kW/ft, four assembly quadrants (one full assembly) or 0.46% of the core, are predicted to fail due to violation of the centerline melt criteria.

The post-scram analysis uses boron injection from the High Pressure Safety Injection (HPSI) System to mitigate the return to power. The HPSI pumps inject borated water from the Refueling Water Storage Tank (RWST) into the RCS. The previous analysis used a non-conservative value for HPSI flow as reported by Licensee Event Report (LER) 97-023-00.[7] The revised analysis corrects this by using a value for HPSI flow that includes expected pump degradation.

A summary of the Post-Scram analysis results is contained in Table 2.

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[7] J. A. Price letter to the NRC, Millstone Nuclear Power Station, Unit No. 2 Licensee Event Report (LER) 97-023-00, dated July 14, 1997.

Table 2
Post-Scram Steam Line Break Analysis Summary
(FSAR Table 14.1.5.2-6)

Initial Power Level	Offsite Power Available	Break Location	Maximum Post-Scram Return to Power (MW)	MDNBR	Maximum LHGR (kW/ft)	Fuel Failure (% of Core due to centerline melt)
HFP	No	Outside Containment	207.5	1.71	17.96	0.0
HFP	Yes	Outside Containment	378.0	2.28	24.27	0.5
HZP	No	Outside Containment	182.9	1.89	15.76	0.0
HZP	Yes	Outside Containment	343.5	2.37	23.47	0.3

Core Acceptance Criteria

The revised Millstone Unit No. 2 main steam line break analysis indicates that the fuel centerline temperature and the departure from nucleate boiling fuel design limits will be exceeded. This has been previously reported to the NRC in LER 98-007-00.[8]

Standard Review Plan (SRP) 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Rev. 2 - July 1981, provides acceptance criteria for main steam line break analysis. The criteria for core damage is presented below.

> The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.2). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

The limited fuel damage that is now predicted in the revised Millstone Unit No. 2 main steam line break analysis meets this criteria. (Note: Millstone Unit No. 2 is not committed to the requirements contained in the Standard Review Plan.)

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[8] J. A. Price letter to the NRC, "Millstone Nuclear Power Station, Unit No. 2, Licensee Event Report 98-007-00, Reanalysis of Main Steam Line Break Indicates Possible Fuel Failures, dated May 14, 1998.

Radiological Consequences Analyses

The significant changes to the radiological consequences analyses are listed below.

1. An increase in allowed control room in-leakage, from 100 SCFM to 130 SCFM, when the Control Room Emergency Ventilation System is operating in the recirculation/filtration mode. This increase will provide additional operational flexibility to address expected minor system degradation over time.

2. A limited amount of fuel failure is predicted in the revised steam line break analysis (no fuel failure is predicted in the current analysis of record). Credit is being taken for the proposed reduction in allowable primary to secondary leakage to 0.035 gpm per steam generator.

3. Credit is now being taken for removing radioactive iodines by the Containment Spray System (CSS) from the post-LOCA atmosphere (iodine scrubbing is not credited in the current analysis of record).

4. The addition of backleakage from the containment sump to the RWST has been included in the off-site and control room LOCA dose consequences analyses.

5. Millstone Unit No. 1 design basis accidents, loss of coolant and main steam line break, will no longer be evaluated for impact on Millstone Unit No. 2 control room habitability. This credits the decision to decommission Millstone Unit No. 1.[9]

6. The revised determination of the radiological consequences of a main steam line break and the design basis accident (LOCA) uses the thyroid dose conversion factors found in ICRP 30. This guide has been used by the NRC in other applications for at least the past 5 years.

The results of the radiological consequences for a main steam line break (MSLB) and the design basis loss of coolant accidents (LOCAs) are presented below.

Radiological Consequences of a Main Steam Line Break

The determination of the radiological consequences of a main steam line break have been revised to be consistent with the revised main steam line break analyses. The radiological consequences of a main steam break outside of containment bound a main steam line break inside of containment.

[9] B. D. Kenyon letter to the NRC, "Millstone Nuclear Power Station, Unit No. 1 Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor," dated July 21, 1998.

Two cases were evaluated for the radiological consequences. The first case is associated with the post-scram main steam line break outside of containment from a full power condition with off-site power available. This case resulted in a fuel failure, due to centerline melt, of 0.46%. Table 3 summarizes the radiological consequences of this case.

Table 3
Summary of Millstone 2 MSLB Accident Doses
(0.46% Melted Fuel)
(FSAR Table 14.1.5.3-2)

Location	Thyroid (rem)	Whole Body (rem)	Beta (rem)
EAB	4.8	0.06	N/A
LPZ	2.3	0.02	N/A
Control Room	29	0.03	0.5

The second case assumes a pre-accident iodine spike occurs. Table 4 summarizes the radiological consequences of this case.

Table 4
Summary of Millstone 2 MSLB Accident Doses
(Pre-accident Iodine Spike)
(FSAR Table 14.1.5.3-3)

Location	Thyroid (rem)	Whole Body (rem)	Beta (rem)
EAB	0.935	0.010	N/A
LPZ	0.176	0.002	N/A
Control Room	5.314	0.003	0.039

The radiological consequences for a main steam line break do not exceed the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) dose limits of 10CFR100 (300 rem thyroid and 25 rem whole body). The dose to the Control Room Operators does not exceed the 10CFR50, Appendix A, General Design Criteria (GDC) 19 limits of 30 rem thyroid, 5 rem whole body, and 30 rem to the skin.

Radiological Consequences of the Design Basis Loss of Coolant Accidents

The determination of the radiological consequences of the design basis LOCA for Millstone Unit No. 2 has been revised to include the effects of backleakage from the containment sump to the RWST. In addition, credit is now being taken for removing radioactive iodines by the CSS from the post-LOCA atmosphere (iodine scrubbing is not credited in the current analysis of record).

Table 5 summarizes the off-si' radiological consequences of the design basis LOCA at Millstone Unit No. 2.

Table 5
Summary of Off-Site Doses for Loss of Coolant Accident
(FSAR Table 14.8.4-2)

Location	Thyroid (rem)	Whole Body (rem)
EAB	45.8	2.46
LPZ	21.9	0.941

Table 6 summarizes the radiological consequences of the design basis LOCAs at Millstone Unit No. 2 or Unit No. 3 to the Millstone Unit No. 2 Control Room Operators.

Table 6
Summary of Dose to Millstone Unit No. 2 Control Room Operators
for Loss of Coolant Accidents
(FSAR Table 14.8.4-5)

Release	Thyroid (rem)	Whole Body (rem)	Beta Skin Dose (rem)
Millstone Unit No. 2 LOCA	25.8	0.718	2.29
Millstone Unit No. 3 LOCA	21.5	1.484	14.0

The radiological consequences for a LOCA at Millstone Unit No. 2 do not exceed the EAB and LPZ dose limits of 10CFR100. The dose to the Control Room Operators from a LOCA at Millstone Unit No. 2 or Unit No. 3 does not exceed the limits specified in GDC 19.

<u>Technical Specification Changes</u>

Changes to the Technical Specifications are necessary to ensure the validity of the revised analyses. These changes, along with additional changes not related to the revised analyses, are discussed below.

Technical Specification 3.3.2.1

1. The spelling of particulate and assumed will be corrected in Surveillance Requirement (SR) 4.3.2.1.4 by replacing "aprticulate" with "particulate" and "asumed" with "assumed."

2. Page 3/4 3-11 was previously revised by License Amendment No. 49.[10] This amendment number will be added to the bottom of the page.

Technical Specification 3.4.6.2

1. The maximum allowable value of primary to secondary leakage will be reduced to 0.035 gpm per steam generator to be consistent with the new radiological assessment of the potential control room operator exposure following a main steam line break outside of containment. The total primary to secondary leakage limit of 1.0 gpm will be deleted since the per steam generator limit is more restrictive.

2. SR 4.4.6.2 will be renumbered as SR 4.4.6.2.1 to allow the addition of a new SR. In addition, the wording of SR 4.4.6.2.1 will be changed to clarify that the water inventory balance is used to verify compliance with the identified and unidentified leakage limits. Pressure boundary leakage would first show up as unidentified leakage during performance of SR 4.4.6.2.1. Further investigation, (plant walkdown) would be necessary to classify the unidentified leakage as pressure boundary leakage. This is consistent with established plant practices to detect pressure boundary leakage.

3. SR 4.4.6.2.2 will be added. This new SR will require the primary to secondary leakage to be verified within limits at least once per 72 hours. This limit has been indirectly verified by performance of the RCS water inventory balance of SR 4.4.6.2. The addition of this new SR will require a specific determination of primary to secondary leakage.

   An exception to Specification 4.0.4 is included with this new SR to allow entry into Mode 4. Prior to entry into Mode 4, the RCS temperature and pressure will not be of sufficient magnitude to provide the driving force for significant primary to secondary leakage. Therefore, even though primary to secondary leakage is possible, it is not expected. The determination of primary to secondary leakage must be made prior to entering Mode 3.

Technical Specification 3.4.8

1. The words "of gross specific activity" will be added to Limiting Condition for Operation (LCO) 3.4.8.b, Action Statement c, and Action Statement d. This proposed change will clarify that the 100/E-bar limit is associated with gross radioactivity, not just iodine activity.

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[10] R. W. Reid letter from the NRC, "Millstone Nuclear Power Station, Unit No. 2 License Amendment No. 49," dated March 1, 1979.

2. The title of the second column of Table 4.4-2 will be changed from "Minimum Frequency" to "Sample and Analysis Frequency." This is not a technical change.

3. Table 4.4-2 - Radiochemical Analysis for E-Bar Determination

   a. The word "Analysis" will be added to the title for clarity. This is not a technical change.

   b. A footnote (*) will be added to specify the power history requirements for the determination of E-Bar. The proposed change is consistent with NUREG - 0212 and NUREG - 1432.

      The footnote will also specify that the provisions of Specification 4.0.4 are not applicable. This will allow entry into Mode 1, without determining the value of E-Bar, assuming that the power history requirements will not be met until after Mode 1 is entered. This will normally only apply following an extended shutdown.

      The failure to perform an E-Bar determination prior to obtaining the necessary power history has resulted in a violation of Technical Specifications as reported in LER 97-022-00 dated July 9, 1997.[11]

4. Table 4.4-2 - Isotopic Analysis for Iodine Including I-131, I-133, and I-135

   a. The sample requirement will be expanded to include the LCO requirement for 100/E-Bar. Minor wording changes will also be made to be consistent with the proposed change to the LCO wording.

Technical Specification 3.6.2.1

1. The revised radiological assessment calculation for the design basis LOCA at Millstone Unit No. 2 credits iodine removal from the containment atmosphere by the CSS. This will require a reduction in the allowed outage time (AOT) of one containment spray train from seven days to seventy two hours. This AOT is consistent with NUREG-0212 and NUREG-1432.

Technical Specification 3.6.5.1

1. The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.6.5.1.d.1 will be changed from $\leq$ 6 inches water gauge to $\leq$ 2.6 inches water gauge. The current value of 6 inches water

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[11] J. A. Price letter to the NRC, Millstone Nuclear Power Station, Unit No. 2 Licensee Event Report (LER) 97-022-00, dated July 9, 1997.

gauge is a generic value. The proposed more restrictive value is a plant specific value.

Technical Specification 3.7.6.1

1. The Control Room Emergency Ventilation System is composed of two separate trains. The proposed wording changes to replace "systems" with "trains," "system" with "train," "air clean – up" with "ventilation," and "control air conditioning" with "control room emergency ventilation" will standardize the terminology used throughout this specification.

2. The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.7.6.1.e.1 will be changed from ≤ 6 inches water gauge to ≤ 3.4 inches water gauge. The current value of 6 inches water gauge is a generic value. The proposed more restrictive value is a plant specific value.

3. SR 4.7.6.1.e.2 will be expanded to clarify that the test of the capability of the Control Room Emergency Ventilation Trains to switch to the recirculation mode is performed with the trains initially operating in the normal mode and the smoke purge mode of operation. These are the only operating modes that the Control Room Emergency Ventilation System can be in before receiving a recirculation actuation signal.

   The failure to test that a recirculation actuation signal overrides the smoke purge actuation signal was identified in NRC Inspection Report 50-336/95-201[12] as Deficiency 95-201-02. The surveillance procedure has already been modified to address this issue. The proposed change will establish the initial conditions necessary for verification of Control Room Emergency Ventilation System operation.

4. The value of allowable control room air in-leakage specified in SR 4.7.6.1.e.3 will be increased from 100 SCFM to 130 SCFM. This is consistent with the recently revised control room radiological analyses for the design basis accidents. The proposed increase will provide additional operational flexibility to address expected minor system degradation over time.

Technical Specification 3.9.15

1. The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.9.15.d.1 will be changed from ≤ 6 inches water gauge to ≤ 2.6 inches water gauge. The current value of 6 inches water gauge

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[12] R. L. Spessard (NRC) letter to NNECO, NRC Inspection Report 50-336/95-201, Millstone Nuclear Power Station Unit 2, Restart Assessment Team Inspection, dated July 21, 1995.

is a generic value. The proposed more restrictive value is a plant specific value.

Technical Specification Bases

The Bases of the applicable Technical Specifications will be revised to reflect the proposed changes.

Safety Summary

Analyses Changes

The main steam line break analyses and the determinations of the radiological consequences of the main steam line break and loss of coolant accident have been revised. A brief summary of the significant changes to the main steam line break analyses and the radiological consequences of the main steam line break and loss of coolant accident is presented below.

1. The limited fuel failure following a main steam line break outside containment results in an increase in the calculated radiological consequences both off-site and in the control room. To limit the consequences to the Millstone Unit No. 2 Control Room Operators following a main steam line break outside containment, the maximum allowable steam generator tube leakage will be reduced to 0.035 gpm per steam generator.

2. Credit will now be taken for iodine removal from the containment atmosphere by the CSS. The use of the CSS for iodine removal has not been previously approved by the NRC.

3. The proposed increase to the allowable control room in-leakage will provide additional operational flexibility to address expected minor system degradation over time. The increase in the allowable control room in-leakage will result in an increase in the calculated control room dose to the Control Room Operators.

4. The addition of the dose consequences from containment sump backleakage to the RWST into the determination of off-site and control room LOCA radiological consequences, increases the consequences of previously evaluated accidents.

5. Credit will be taken in the main steam line break analyses for the recently installed cavitating venturis in the Auxiliary Feedwater System.

6. Credit will be taken for the low RCS flow reactor trip for the pre-scram inside containment main steam line break analysis. This equipment will be qualified for the expected containment environment following a main steam line break inside containment and will be added to the Environmental Qualification Master List.

7. Millstone Unit No. 1 design basis accidents, loss of coolant and main steam line break, will no longer be evaluated for impact on Millstone Unit No. 2 control room habitability. This credits the decision to decommission Millstone Unit No. 1.

The revised main steam line break analyses and the revised determinations of the radiological consequences of the main steam line break and loss of coolant accident analyses take credit for equipment not previously assumed in the analyses, and for plant or equipment operating restrictions not currently contained in the Technical Specifications. However, the changes proposed in this License Amendment Request are consistent with the revised analyses. In addition, the results of the new analyses meet the guidance contained in SRP 15.1.5, SRP 15.6.5, and the limits of 10CFR100 and GDC 19. Therefore, there will be no adverse impact on public health and safety as a result of the revised analyses.

Technical Specification Changes

Technical Specification Non-Technical Changes

The minor editorial and non-technical changes to correct spelling (Technical Specification 3.3.2.1), modify the title of a table column (Technical Specification 3.4.8), clarify the type of measurement performed (Technical Specification 3.4.8), and establish consistent terminology (Technical Specification 3.7.6.1) 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 3.4.6.2

The reduction in the maximum allowable value of primary to secondary leakage per steam generator is consistent with the new radiological assessment of the potential control room operator exposure following a main steam line break outside of containment. The wording change to SR 4.4.6.2.1 will clarify that the water inventory balance is used to verify compliance with the identified and unidentified leakage limits. Pressure boundary leakage would first show up as unidentified leakage during performance of SR 4.4.6.2.1. Further investigation, (plant walkdown) would be necessary to classify the unidentified leakage as pressure boundary leakage. This is consistent with established plant practices to detect pressure boundary leakage.

The addition of the new SR 4.4.6.2.2 will address the primary to secondary leakage limit. The new SR will include an exception to Technical Specification 4.0.4 that will allow the determination of primary to secondary leakage to be deferred until after Mode 4 is entered. Even though verification of compliance with the primary to secondary limit will not be done prior to entering Mode 4, the limit is still expected to be met.

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 3.4.8

The addition of the words "of gross specific activity" to the LCO, Action Statements, and SR will clarify what the E-Bar limit applies to. This is consistent with the Technical Specification Definition (1.20) for E-Bar.

The addition of a footnote (*) to state the power history requirements for the determination of E-Bar will ensure that the necessary plant conditions are established prior to performing the analysis. This will not affect the E-Bar LCO limit or the requirement to perform the analysis. The proposed change is consistent with NUREG - 0212 and NUREG - 1432.

The footnote will also specify that the provisions of Specification 4.0.4 are not applicable. This will allow entry into Mode 1, without determining the value of E-Bar, assuming that the power history requirements will not be met until after Mode 1 is entered. This will normally only apply following an extended shutdown.

The Isotopic Analysis for Iodine (including I-131, I-133, and I-135) sample requirement will be expanded to include the LCO requirement for 100/E-Bar. This is consistent with the requirements of Action Statement d. This change will expand the sampling requirement for iodine. Minor wording changes will also be made to be consistent with the proposed changes to the LCO wording.

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 3.6.2.1

The revised radiological assessment calculation for the Millstone Unit No. 2 design basis LOCA credits iodine removal from the containment atmosphere by the CSS. This will require a reduction in the AOT of one containment spray train from seven days to seventy two hours. This AOT is consistent with NUREG-0212 and NUREG-1432. This will help ensure that plant equipment assumed in the safety analyses will be available. This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no adverse impact on public health and safety.

Technical Specification 3.6.5.1

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.6.5.1.d.1 will be changed from a generic value ($\leq$ 6 inches water gauge) to a plant specific value ($\leq$ 2.6 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no adverse impact on public health and safety.

Technical Specification 3.7.6.1

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.7.6.1.e.1 will be changed from a generic value ($\leq$ 6 inches water gauge) to a plant specific value ($\leq$ 3.4 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation.

SR 4.7.6.1.e.2 will be expanded to clarify that the test of the capability of the Control Room Emergency Ventilation Trains to switch to the recirculation mode is performed with the trains initially operating in the normal mode and the smoke purge mode of operation. This will not affect the requirement that the trains be capable of switching to the recirculation mode.

The value of allowable control room air in-leakage specified in SR 4.7.6.1.e.3 will be increased from 100 SCFM to 130 SCFM. This is consistent with the recently revised control room radiological analysis for the design basis accidents. The proposed increase will provide additional operational flexibility to address expected minor system degradation over time. This increase is supported by the new analyses.

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 3.9.15

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.9.15.d.1 will be changed from a generic value ($\leq$ 6 inches water gauge) to a plant specific value ($\leq$ 2.6 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no adverse impact on public health and safety.

Technical Specification Bases

The Bases of the applicable Technical Specifications will be revised to reflect the proposed changes.

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 since the changes are consistent with the revised analyses, and the appropriate acceptance criteria are met for the revised analyses. 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
Control Room Ventilation System
Significant Hazards Consideration

September 1998

Proposed Revision to Technical Specifications
Control Room Ventilation System
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.

   Analyses Changes

   The main steam line break analyses and the determinations of the radiological consequences of the main steam line break and loss of coolant accident have been revised. A brief summary of the significant changes to the main steam line break analyses and the radiological consequences of the main steam line break and loss of coolant accident is presented below.

   1. The limited fuel failure following a main steam line break outside containment results in an increase in the calculated radiological consequences both off-site and in the control room. To limit the consequences of a main steam line break outside containment, the Technical Specification allowed steam generator tube leakage will be reduced to 0.035 gpm per steam generator.

   2. Credit will now be taken for iodine removal from the containment atmosphere by the Containment Spray System (CSS). The use of the CSS for iodine removal has not been previously approved by the NRC.

   3. The proposed increase to the allowable control room in-leakage will provide additional operational flexibility to address expected minor system degradation over time. The increase in the allowable control room in-leakage will result in an increase in the calculated dose to the Control Room Operators.

   4. The addition of the dose consequences from containment sump backleakage to the Refueling Water Storage Tank (RWST) has been included in the off-site and control room loss of coolant accident (LOCA) analyses increases the consequences of previously evaluated accidents.

The containment sump backleakage into the RWST results in sump water entering the RWST when the RWST is at its minimum level. The RWST will become a radioactive source and contribute a shine dose to the surrounding areas. The increase in dose rates onsite will not prevent operators from remaining in the control room or from accessing equipment needed to mitigate the accident.

All piping and valves associated with RWST backleakage are located in a harsh radiation area. Backflow from the sump might increase dose rates in the area where these components are located. Additional dose contributions, where they occur, do not adversely impact the environmental qualification of the vital equipment located there. All vital equipment would continue to perform its safety function.

5. Credit will be taken in the main steam line break analyses for the recently installed cavitating venturis in the Auxiliary Feedwater System. However, this will not change the amount of fuel failure. Therefore, credit for this equipment will not impact the radiological consequences of a main steam line break.

6. Credit will be taken for the Reactor Coolant System (RCS) low flow reactor trip for the pre-scram inside containment main steam line break analysis. This equipment will be qualified for the expected containment environment following a main steam line break inside containment and will be added to the Environmental Qualification Master List.

7. Millstone Unit No. 1 design basis accidents, loss of coolant and main steam line break, will no longer be evaluated for impact on Millstone Unit No. 2 control room habitability. This credits the decision to decommission Millstone Unit No. 1.[1]

The revised main steam line break analyses and the revised determinations of the radiological consequences of the main steam line break and design basis LOCA analyses take credit for equipment not previously assumed in the analyses, and for plant or equipment operating restrictions not currently contained in the Technical Specifications. The changes to the analyses will not adversely affect the probability of an accident previously evaluated, but the revised analyses results do indicate that the consequences of an accident

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[1] B. D. Kenyon letter to the NRC, "Millstone Nuclear Power Station, Unit No. 1 Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor," dated July 21, 1998.

previously evaluated will increase. Specifically, the following changes cause an increase in the consequences of an accident previously evaluated.

1. The increase in allowable control room in-leakage from 100 SCFM to 130 SCFM when the Control Room Emergency Ventilation System is operating in the recirculation/filtration mode.

   The dose to the Control Room Operators from a Millstone Unit No. 2 LOCA increased from 9.25 to 25.8 rem to the thyroid and from 0.205 to 2.29 rem to the skin. The dose to the whole body decreased. (Both low wind speed and high wind speed release conditions were analyzed. The low wind speed condition bounds the high wind speed condition.) The dose to the Control Room Operators from a Millstone Unit No. 3 LOCA increased from 2.67 to 14 rem to the skin and from 0.209 to 1.484 rem to the whole body. The dose to the thyroid decreased. The doses to the Control Room Operators from either a Millstone Unit No. 2 or Unit No. 3 LOCA remain below the GDC 19 criteria of 30 rem thyroid, 5 rem whole body and 30 rem to the skin.

   The new calculated doses to the Millstone Unit No. 2 Control Room Operators from a main steam line break outside containment are 29 rem thyroid, 0.03 rem whole body and 0.5 rem skin. The doses to the Millstone Unit No. 2 Control Room Operators are below the GDC 19 criteria of 30 rem thyroid, 5 rem whole body, and 30 rem to the skin. (Note: The dose to the Control Room Operators from a main steam line break was not previously evaluated because fuel failure was not predicted to occur.)

2. The limited fuel failure that is predicted in the revised main steam line break analyses.

   Previously, the radiological consequences of a main steam line break were not determined and were not presented in the FSAR because fuel failure was not predicted to occur. Because of the predicted limited fuel failure for the main steam line break outside of containment, the radiological consequences were analyzed. The results to the Exclusion Area Boundary (EAB) are 4.8 rem thyroid and 0.06 rem whole body. The results to the Low Population Zone (LPZ) are 2.3 rem thyroid and 0.02 rem whole body. To meet the dose acceptance criteria to the Millstone Unit No. 2 Control Room Operators, the maximum allowable Technical Specification primary to secondary leak rate is being reduced to 0.035 gpm per steam generator. The results to the Millstone Unit No. 2 Control Room Operators are 29 rem thyroid, 0.03 rem whole body and 0.5 rem skin. The main steam line break outside containment is the limiting accident for the Millstone Unit No. 2 Control Room Operators. However, the dose consequences of a main steam line break are less than the 10CFR100 limits off-site of 300 rem thyroid and 25 rem whole body, and the doses to the Millstone Unit No. 2 Control Room Operators are below the GDC 19 criteria of 30 rem thyroid, 5 rem whole body, and 30 rem to the skin.

3. Taking credit for the low RCS flow reactor trip for the pre-scram inside containment main steam line break analysis.

   Previous analyses did not credit the low RCS flow reactor trip in a harsh environment. This credits the low flow trip in a manner not previously reviewed by the NRC for Millstone Unit No. 2. Without credit for this reactor trip, the predicted fuel failure for steam line breaks inside containment would be higher.

4. Taking credit for the removal of radioactive iodine from the containment atmosphere by containment spray.

   Previous analyses did not rely on the spray function to reduce iodine concentration in the post-accident atmosphere inside containment. This adds a mitigation function to the CSS that has not been previously reviewed by the NRC for Millstone Unit No. 2. Without credit for the removal of iodine, the predicted dose consequences following a LOCA would be higher.

5. The addition of sump backleakage to the RWST during a LOCA.

   The resultant dose contribution to the LPZ from RWST backleakage is 1.487 rem thyroid and 0.11 rem whole body. The total dose to the LPZ from a design basis LOCA is 21.86 rem thyroid and 0.941 rem whole body. The dose is well below the 10CFR100 limits of 300 rem thyroid and 25 rem whole body. The dose to the EAB was not affected because leakage into the RWST does not start until 25.45 hours post-LOCA and the EAB is a 2-hour dose.

   The resultant dose contribution to the Millstone Unit No. 2 Control Room Operators from RWST backleakage is 3.75 rem thyroid, 0.017 rem whole body and 0.296 to the skin. The total dose to the Millstone Unit No. 2 Control Room Operators from the LOCA is 25.8 rem thyroid, 0.718 rem whole body and 2.29 rem to the skin. These doses are below the GDC 19 limits of 30 rem thyroid and skin, and 5 rem whole body.

The analyses results meet the guidance contained in SRP 15.1.5, SRP 15.6.5, and the limits of 10CFR100 and GDC 19. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification Changes

Technical Specification Non-Technical Changes

The minor editorial and non-technical changes to correct spelling (Technical Specification 3.3.2.1), modify the title of a table column (Technical Specification 3.4.8), clarify the type of measurement performed (Technical Specification 3.4.8), and establish consistent terminology (Technical Specification 3.7.6.1) 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 significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.6.2

The reduction in the maximum allowable value of primary to secondary leakage per steam generator is consistent with the new radiological assessment of the potential control room operator exposure following a main steam line break outside of containment. The wording change to SR 4.4.6.2.1 will clarify that the water inventory balance is used to verify compliance with the identified and unidentified leakage limits. Pressure boundary leakage would first show up as unidentified leakage during performance of SR 4.4.6.2.1. Further investigation, (plant walkdown) would be necessary to classify the unidentified leakage as pressure boundary leakage. This is consistent with established plant practices to detect pressure boundary leakage.

The addition of the new SR 4.4.6.2.2 will address the primary to secondary leakage limit. The new SR will include an exception to Technical Specification 4.0.4 that will allow the determination of primary to secondary leakage to be deferred until after Mode 4 is entered. Even though verification of compliance with the primary to secondary limit will not be done prior to entering Mode 4, the limit is still expected to be met.

The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.8

The addition of the words "of gross specific activity" to the Limiting Condition for Operation (LCO), Action Statements, and SR will clarify what the E-Bar limit applies to. This is consistent with the Technical Specification Definition (1.20) for E-Bar.

The addition of a footnote (*) to state the power history requirements for the determination of E-Bar will ensure that the necessary plant conditions are established prior to performing the analysis. This will not affect the E-Bar LCO limit or the requirement to perform the analysis. The proposed change is consistent with NUREG - 0212 and NUREG - 1432.

The footnote will also specify that the provisions of Specification 4.0.4 are not applicable. This will allow entry into Mode 1, without determining the value of E-Bar, assuming that the power history requirements will not be met until after Mode 1 is entered. This will normally only apply following an extended shutdown.

The Isotopic Analysis for Iodine (including I-131, I-133, and I-135) sample requirement will be expanded to include the LCO requirement for 100/E-Bar. This is consistent with the requirements of Action Statement d. This change will expand the sampling requirement for iodine. Minor wording changes will also be made to be consistent with the proposed changes to the LCO wording.

The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.6.2.1

The revised radiological assessment calculation for the design basis accident credits iodine removal from the containment atmosphere by the CSS. This will require a reduction in the allowed outage time (AOT) of one containment spray train from seven days to seventy two hours. This AOT is consistent with NUREG-0212 and NUREG-1432. This will help ensure that plant equipment assumed in the safety analyses will be available. This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.6.5.1

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.6.5.1.d.1 will be changed from a generic value

(≤ 6 inches water gauge) to a plant specific value (≤ 2.6 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.7.6.1

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.7.6.1.e.1 will be changed from a generic value (≤ 6 inches water gauge) to a plant specific value (≤ 3.4 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation.

SR 4.7.6.1.e.2 will be expanded to clarify that the test of the capability of the Control Room Emergency Ventilation Trains to switch to the recirculation mode is performed with the trains initially operating in the normal mode and the smoke purge mode of operation. This will not affect the requirement that the trains be capable of switching to the recirculation mode.

The value of allowable control room air in-leakage specified in SR 4.7.6.1.e.3 will be increased from 100 SCFM to 130 SCFM. This is consistent with the recently revised control room radiological analysis for the design basis accidents. The proposed increase will provide additional operational flexibility to address expected minor system degradation over time. This increase is supported by the new analysis.

The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.9.15

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.9.15.d.1 will be changed from a generic value (≤ 6 inches water gauge) to a plant specific value (≤ 2.6 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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 since the changes are consistent with the revised analyses, and the appropriate acceptance criteria are met for the revised analyses. 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. Also, the response of the plant and the operators following these accidents is unaffected by the change. 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.

   Analyses Changes

   The acceptance criteria for a main steam line break in the SRP 15.1.5 does not exclude the prediction of fuel failure. Instead, the SRP requires that "Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling." The limited fuel failure that is now predicted in the revised main steam line break analyses meets this acceptance criterion. In addition, the RCS low flow reactor trip that is now being credited to function in a harsh environment to limit fuel failure is already required to be operable by Technical Specifications.

   The maximum allowable primary to secondary leakage is being reduced to 0.035 gpm per steam generator. This reduction in allowable leakage is credited in the analyses for dose consequences for main steam line break outside containment. In addition, the revised dose consequences for the main steam line break assumes a control room in-leakage of 130 SCFM. The dose consequence analyses, using these new limits, show that the dose acceptance criteria are met for the main steam line break. Therefore, the fact that a limited fuel failure is now predicted does not involve a significant reduction in the margin of safety.

   The revised dose consequences for the design basis accidents assumes a control room in-leakage of 130 SCFM. In addition, iodine removal by the CSS, which is already required to be operable by Technical Specifications, is assumed. The acceptance criteria for the dose consequences of the design basis accidents to the EAB, LPZ and the control room personnel is met in the revised analyses. Therefore, the revisions to the dose consequence analyses for the design basis accidents do not involve a significant reduction in the margin of safety.

Technical Specification Changes

The proposed changes will correct spelling and terminology errors, reduce the maximum allowable primary to secondary leakage, add a new surveillance requirement, modify surveillance requirements for RCS specific activity, reduce the allowed outage time for a containment spray train, reduce the allowed pressure drop across the control room and enclosure building HEPA filters, and increase the control room maximum allowed in-leakage. 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 only adverse impact of the proposed changes is that the dose consequences following an accident may increase. However, the revised analyses show that the acceptance criteria for the accident analyses are met. Therefore, based on the responses above, the proposed changes are deemed safe.

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. The changes proposed herein to add a new surveillance requirement to verify primary to secondary leakage and to reduce the allowable pressure drop across various ventilation filters are enveloped by example (ii), a change that constitutes an additional limitation, restriction, or control not presently included in the 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.