Northeast
Utilities System

107 Selden Street, Berlin, CT 06037

Northeast Utilities Service Company
P.O. Box 270
Hartford, CT 06141-0270
(203) 665-5000

December 14, 1994

Docket No. 50-423
B15028

Re: 10CFR50.90

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

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specifications
Supplementary Leak Collection and Release System

Introduction

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its Operating License, NPF-49, by incorporating the changes into the Technical Specifications of Millstone Unit No. 3. Specifically, NNECO proposes to revise the Millstone Unit No. 3 Technical Specifications by:

1. Increasing the upper bound of the overall containment integrated leakage rate required by Technical Specification 3.6.1.2.a from 0.3 wt. % per day to 0.65 wt. % per day of the containment air per 24 hours at design basis pressure.

2. Revising Technical Specification 4.6.6.1.d.3 by providing more margin with respect to the drawdown time for secondary containment vacuum.

3. Revising Bases Section 3/4.7.9 to reflect the above changes.

Background

NNECO notified the NRC Staff of a design deficiency in the auxiliary building filter system (ABFS) in License Event Report

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(LER) 93-014-00.[1] In this LER, NNECO describes corrective actions which included design changes to the ABFS. Upon completion of the modifications, NNECO performed an engineered safety feature (ESF)/loss of power (LOP) test to verify system operability. During the performance of the test, the 'B' train ABFS fan did not start until 90 seconds after a safety injection signal (SIS). This failure rendered the ABFS and the supplementary leak collection and release system (SLCRS) inoperable. The engineering review of this failure identified additional single-failure vulnerabilities with the SLCRS/ABFS instrumentation and controls. Several design/operational changes were implemented during the last refueling outage (Cycle 4 Refueling Outage) to correct the identified deficiencies. In addition, a series of integrated tests were performed under various normal operating and simulated failure modes, to demonstrate that the ABFS and SLCRS would achieve drawdown to a negative pressure of 0.4 inches water gauge as measured at the 24'-6" elevation in the auxiliary building within 120 seconds following an accident signal (this time includes the diesel generator start and load time of approximately 10 seconds).

In a letter dated November 4, 1993,[2] NNECO submitted a proposed license amendment which requested that: (1) the time to draw a negative pressure of 0.4 inches water gauge as measured at the 24-'6" elevation of the auxiliary building from 60 seconds to 120 seconds, and (2) the containment integrated leakage rate at the design basis pressure be reduced from 0.65 wt.%/day to 0.30 wt. %/day. This amendment was approved by the NRC on December 8, 1993.[3]

To enhance flexibility and to increase reliability of the SLCRS and the ABFS, NNECO proposes to revise the current surveillance requirement for the drawdown time for the secondary containment

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(1) S. E. Scace letter to the U.S. Nuclear Regulatory Commission, "Facility Operating License No. NPF-49, Docket No. 50-423, Licensee Event Report 93-014-00," dated September 30, 1993.

(2) J. F. Opeka letter to the U.S. Nuclear Regulatory Commission, "Proposed Revision to the Technical Specifications, Supplementary Leak Collection and Release System," dated November 4, 1993.

(3) V. L. Rooney letter to J. F. Opeka, "Issuance of Amendment (TAC No. M87216)," dated December 8, 1993.

vacuum. Specifically, the revised surveillance requirement will ensure that a minimum negative pressure of 0.1 inches water gauge will be achieved within one minute and the final required negative pressure of 0.4 inches water gauge will be achieved within the next two minutes. This will allow NNECO to deviate from the provisions of the Standard Review Plan (SRP) 6.5.3 which requires that a negative pressure of 0.25 inches water gauge be attained within the secondary containment boundary in less than one minute. The proposed amendment will permit NNECO to assume that no exfiltration will occur through the secondary containment boundary.

Description of Proposed Technical Specification Changes

NNECO proposes to revise the Millstone Unit No. 3 Technical Specifications as follows:

1. Section 3.6.1.2.a, Containment Leakage

   The containment overall integrated leakage rate, L_a is being changed from 0.3 wt. % to 0.65 wt. % of the containment air per 24 hours at the design basis pressure. This will restore the containment overall integrated leakage rate (L_a) acceptance criterion that was modified as a result of the SLCRS/ABFS drawdown time issue in October 1993. The proposed change to 0.65 wt. % per day makes 'L_a' consistent with the assessment included in the February 26, 1990,[4] submittal.

2. Section 4.6.6.1.d.3, SLCRS - Surveillance Requirement

   Surveillance Requirement 4.6.6.1.d.3 is being revised to provide a more restrictive requirement for the first minute and a greater margin with respect to the final drawdown time for the secondary containment vacuum. Specifically, the revised surveillance requirement will ensure that a minimum negative pressure of 0.1 inches water gauge will be achieved within one minute and the final required negative pressure of 0.4 inches water gauge within the next two minutes. The revised Surveillance Requirement 4.6.6.1.d.3 will read as follows:

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(4) E. J. Mroczka letter to the U.S. Nuclear Regulatory Commission, "Proposed Revision to Technical Specifications, Containment Pressure," dated February 26, 1990.

"Verifying that each system produces a negative pressure of greater than or equal to 0.1 inch Water Gauge in the Auxiliary Building at 24'6" elevation within 60 seconds after a start signal and a negative pressure of greater than or equal to 0.4 inches Water Gauge in the Auxiliary Building at 24'6" elevation within the next 120 seconds."

This will allow NNECO to deviate from the provisions of the SRP 6.5.3 which requires the attainment of a negative pressure of 0.25 inches water gauge within the secondary containment boundary in one minute to exclude any exfiltration.

3. Bases Section 3/4.6.6 - SECONDARY CONTAINMENT

Appropriate changes are proposed to Bases Section 3/4.6.6 to reflect the changes proposed to Surveillance Requirement 4.6.6.1.d.3. In addition, a reference to hydrogen recombiner ventilation system alignment after the start of the SLCRS is being deleted from the Bases Section 3/4.6.6.1 (page B3/4 6-6). This change has no impact on the safety.

The marked up technical specification pages are provided in Attachment 2 and the retyped technical specification pages are provided in Attachment 3. These retyped pages reflect the currently issued version of the Millstone Unit No. 3 Technical Specifications.

Safety Assessment

The proposed change to Technical Specification Section 3.6.1.2.a will reinstate a containment leakage rate (L_a) of 0.65 wt. % per day. L_a was changed to 0.3 wt. % per day to compensate for the increased drawdown time (i.e., two minutes) of unfiltered leakage. The following safety assessment addresses the reinstatement of L_a and determines the impact to the Appendix J requirements for Type A, B, and C testing. In addition, the radiological consequence evaluation provided later in this submittal addresses the increase in L_a (i.e., from 0.3 wt.% per day to 0.65 wt.% per day.)

Containment Leakage

Primary containment integrity ensures that the release of radioactivity to the environment will be restricted to those leakage paths and leakage rate values assumed in the safety

analyses. In conjunction with these restrictions, a limitation on the containment leakage rate is imposed to ensure that the overall leakage rate will not exceed the value assumed in the safety analyses. As an added conservatism, 10CFR50, Appendix J requires that the overall integrated leakage rate be further restricted to less than 0.75 L_a during periodic surveillance testing of containment (i.e., Type A test). The Type A test, also known as the Integrated Leakage Rate Test (ILRT), is intended to measure the primary reactor containment overall leakage rate at accident pressure, P_a, through the primary containment and system and components penetrating primary containment. This leakage rate measurement in weight percent of contained air per day represents the leakage rate which would exist under design basis accident (DBA) conditions for the containment in its present state.

On October 12, 1993, Millstone Unit No. 3 successfully conducted the second Type A test in the first 10-year service period. Test results indicated that the "As-Found" and "As-Left" ILRTs passed the technical specification acceptance criteria. The "As-Found" value was 0.1327 weight percent per day and the "As-Left" value was 0.1313 weight percent per day. These values represent 27.2% and 26.9% of the technical specification criterion of 0.4875 wt.%/day (0.75 L_a), based on La equal to 0.65 wt.%/day, respectively. In addition, as of October 9, 1993, the total Type B and C "As-Found" and "As-Left" leakage results were 0.099 wt.% per day, and 0.084 wt. % per day, respectively. These values represent approximately 25.3% and 21.5% of the technical specification limit of 0.39% wt.%/day (0.6 L_a), based on L_a equal of 0.65 wt.%/day, respectively. Correspondingly, the 1993 Type A, B, and C test results indicate that the "As-Found" and "As-Left" result in each test case was below the existing Technical Specification limit of 0.3 wt.%/day. This further demonstrates the overall leakage integrity of the containment and its boundaries.

Based on the relatively low "As-Left" ILRT leakage rate (i.e., 0.1313 wt % per day is well below the existing technical specifications limit of 0.225 wt % per day (0.75 L_a), based on L_a equal to 0.3 wt % per day), which represents the overall containment integrated leakage rate for the containment prior to start-up, there is reasonable assurance that containment integrity will be maintained below the allowable leakage rate limit of 0.65 wt.%/day. In addition, the total Type B and C "As-Left" leakage result of 0.084 wt.%/day (this is well below the existing technical specification limit of 0.18 wt % per day [0.6 L_a], based on L_a equal to 0.3 wt % per day) provides further assurance that

leakage, based on individual penetration, will be maintained within sufficient margin of the leakage limits.

Because the last Type A, B, and C tests were performed under the technical specification limit of 0.65 wt.%/day, the proposed change to restore L_a to 0.65 wt.%/day has no impact to these systems from a leakage allowance perspective. As indicated above, the previous test results met the technical specification leakage limits (based on 0.65 wt.%/day) within sufficient margin and, therefore, would not present any challenge to these leakage limits.

Secondary Containment Pressure

The SLCRS design basis is established by the consequences of the DBA, which is a loss of coolant accident (LOCA). The accident analysis assumes that only one train of the SLCRS and one train of the ABFS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction of the airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from the containment is determined for a LOCA.

The proposed changes to Technical Specifications 4.6.6.1.d.3 will verify that one train of SLCRS in conjunction with the AFBS will produce a negative pressure of greater than or equal to 0.1 inches water gauge in the auxiliary building at 24'-6" elevation within 60 seconds after a start signal (this includes the diesel generator start and load time of approximately 11 seconds) and a negative pressure of greater than or equal to 0.4 inches water gauge in the auxiliary building at 24'-6" elevation within the next 120 seconds. For the purpose of this surveillance, pressure measurements will be taken at the 24'-6" elevation in the auxiliary building. This single location is considered to be adequate and representative of the entire secondary containment due to the large cross-section of the air passages which interconnect the various buildings within the boundary.

With the proposed negative pressure criteria for the secondary containment, it could be assumed that all leakage into the secondary containment is filtered, since a sufficient negative pressure is achieved within one minute and the final required negative pressure is achieved within the next two minutes. This assumption deviates from the provisions of SRP 6.5.3 which recommends a drawdown to a negative pressure of 0.25 inches water

gauge in one minute. The deviation from SRP 6.5.3 is acceptable based on the following considerations:

1. Significant fuel damage is not realistically expected during the first few minutes after a LOCA.

2. Transport time from the reactor core to the outer walls of the containment, through the containment walls and then through the secondary containment to the outer walls of those structures is expected to take several minutes. This is primarily true because the most likely locations of containment leakage (i.e., the areas with many penetrations) are typically located centrally in the surrounding structures.

3. The proposed technical specification and corresponding test criterion will ensure that a minimum negative pressure of 0.1 inch water gauge will be achieved within one minute. Based on previous test (i.e., October 1993 tests) results which showed a fairly consistent drawdown with time between various locations, this negative pressure criterion will ensure that all areas of the building are negative to the outside under the lower wind speed conditions used in postulated accident X/Q calculation assumptions. For those higher wind speed conditions where all surfaces of the secondary containment may not be negative, the negative pressure criterion, plus the trend towards 0.4 inches water gauge negative within the next two minutes, plus the relative leak tightness of the secondary containment required to be maintained to achieve the final required negative pressure will ensure that the volume of secondary containment air that might leak out from one to three minutes is insignificant.

In nearly all probable scenarios, the final required 0.4 inches water gauge pressure is expected to be achieved within one minute. The inability to do so requires a loss of normal power and certain single failures. In all cases, a 0.4 inches water gauge negative pressure is expected to be reached in three minutes.

<u>Radiological Consequences</u>

Appendix A of Section 15.6.5 of the SRP and Regulatory Guide 1.4 provide the guidelines for performing the calculation of radiological consequences for a potential LOCA. These documents establish a set of simplifying nonmechanistic assumptions to be used in the calculation. Utilizing these assumptions permits one

to reach the conclusion specified in the SRP that the distance to the exclusion are a boundary (EAB) and to the low population zone (LPZ) of the site, in conjunction with the engineered safety features of the plant are sufficient to provide reasonable assurance that the total radiological consequences of such an accident will be within the exposure guidelines set forth in 10CFR Part 100.11.

The design basis accidents that are impacted by the proposed change to the containment integrated leak rate and secondary containment drawdown time (i.e., filtration) are the LOCA and the control rod ejection (CRE) accident. In our November 4, 1993, submittal,[5] only the LOCA was reanalyzed since the CRE accident consequences are bounded by the LOCA. The current Final Safety Analysis Report (FSAR) analysis for the CRE accident assumes no secondary containment bypass and assumes L_a = 0.65 wt. % per day. Therefore, no additional evaluation is necessary for the CRE accident. Only the consequences due to a LOCA need to be redetermined.

NNECO has evaluated the proposed changes to Surveillance Requirement 4.6.6.1.d.3 that increases the time to draw a final required negative pressure as measured at the 24'-6" elevation of the auxiliary building in conjunction with the proposed change to reinstate the containment integrated leakage rate of 0.65 wt.% per day to determine the impact on the offsite doses following a LOCA. The calculated radiological doses are, in most cases, less than the previously calculated doses (i.e., EAB and LPZ doses) and are within the 10CFR100 limits.

These are depicted in the Table below.

LOCATION/DOSE TYPE	LIMIT	Feb. 26, 1990 SUBMITTAL	Nov. 4, 1993 SUBMITTAL	CURRENT CALCULATION
EAB 0-2 HR Thyroid EAB 0-2 HR Whole Body	300 REM 25 REM	150 REM 19.5 REM	141 REM 9.4 REM	61 REM 16.7 REM

(5) J. F. Opeka letter to the U.S. Nuclear Regulatory Commission, "Proposed Revision to Technical Specifications, Supplementary Leak Collection and Release System," dated November 4, 1993.

LPZ 30-Day Thyroid	300 REM	31.6 REM	29.8 REM	10.9 REM
LPZ 30-Day Whole Body	25 REM	3.5 REM	1.7 REM	2.8 REM

The assumptions used in the above radiological dose calculations are provided in Attachment 1. Specifically, Attachment 1 includes a discussion on the release pathways and the assumptions associated with the iodine removal from containment atmosphere by the containment spray systems. This iodine removal coefficients are based on the elimination of the existing chemical addition tank (CAT) as the means for sump ph control and replacing it with the trisodium phosphate (TSP) baskets in the sump. The substitution of the CAT with the TSP baskets is scheduled to take place in the next refueling outage. The technical specifications relating to the TSP baskets are being submitted under a separate cover. It is noted that a LOCA at Millstone Unit No. 3 is also one of the bounding accidents for the Millstone Unit No. 3 control room, Millstone Unit No. 2 control room, and the Millstone Technical Support Center habitability analysis. Therefore, the doses for these areas were recalculated and are presented below. The assumptions used for the radiological dose calculations are provided in Attachment 1. It is noted that the Millstone Unit No. 1 control room and the Emergency Operating Facility doses are bounded by the Millstone Unit No. 1 LOCA calculations.

LOCATION/DOSE TYPE	LIMIT	FEB. 26, 1990 SUBMITTAL	CURRENT CALCULATION
MP3 Control Room - Thyroid	30 REM	26 REM	7.9 REM
MP3 Control Room - Whole Body	5 REM	3.05 REM	4.1 REM
MP3 Control Room - Skin	30 REM	24.5 REM	25.5 REM
MP2 Control Room - Thyroid	30 REM	18.4 REM	10.3 REM
MP2 Control Room - Whole Body	5 REM	0.5 REM	2.4 REM
MP2 Control Room - Skin	30 REM	8.3 REM	8.2 REM
MP Tech. Support Center - Thyroid	30 REM	7.4 REM	3.3 REM
MP Tech. Support Center - Whole Body	5 REM	1.4 REM	2.3 REM
MP Tech. Support Center - Skin	30 REM	24.9 REM	29.9 REM

It is noted that the Millstone Unit No. 2 and No. 3 control rooms and Millstone Technical Support Center doses were not recalculated in 1993 (i.e., November 4, 1993, submittal) since EAB/LPZ doses proved that the releases were less than the 1990 submittal. In

summary, all control room and Technical Support Center doses are within the guidelines of General Design Criterion (GDC) 19. Therefore, the proposed changes do not result in an increase in consequences of an accident (i.e., a LOCA) previously analyzed.

Significant Hazards Consideration

In accordance with 10CFR50.92, NNECO has reviewed the attached 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 satisfied. The proposed changes do not involve an SHC because the changes would not:

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

   The proposed changes to Technical Specification 4.6.6.1.d.3 will verify that one train of SLCRS in conjunction with the AFBS will produce a negative pressure of greater than or equal to 0.1 inches water gauge in the auxiliary building at 24'-6" elevation within 60 seconds after a start signal (this includes the diesel generator start and load time of approximately 11 seconds) and a negative pressure of greater than or equal to 0.4 inches water gauge in the auxiliary building at 24'-6" elevation within the next 120 seconds. For the purpose of this surveillance, pressure measurements will be taken at the 24'-6" elevation in the auxiliary building. This single location is considered to be adequate and representative of the entire secondary containment due to the large cross-section of the air passages which interconnect the various buildings within the boundary.

   The test performed during October 1993 confirmed that the SLCRS/ABFS will achieve drawdown to a negative pressure of 0.4 inches water gauge as measured at the 24'-6" elevation in the auxiliary building within 120 seconds following an accident signal. There is a reasonable assurance that the modified criteria for the negative pressure in the secondary containment boundary proposed via the proposed change (i.e., a negative pressure of 0.1 inches in one minute and a negative pressure of 0.4 inches within the next two minutes), can be accomplished in the prescribed time.

   Extension of the time allowed to achieve the final drawdown of

secondary containment from 120 seconds to 180 seconds (these times include the diesel generator start and load time of approximately 11 seconds) will have a negligible impact on heating and cooling. Plant experience has shown that heatup and cooldown of thick-walled concrete structures, such as the Millstone Unit No. 3 auxiliary building, is a relatively slow process. Also, natural convection within the auxiliary building tends to stabilize temperatures. Following an accident signal, ventilation equipment is restarted promptly. Therefore, heatup or cooldown, during short periods while ventilation fans and/or heaters are inactive, is insignificant and can be neglected.

The proposed change to reinstate the containment integrated leakage rate at the design basis pressure from 0.3 wt % per day to 0.65 wt % per day has been evaluated to determine the impact to the Appendix J requirements for Type A, B and C Testing. In addition, the radiological consequence evaluation also addressed the increase in L_a (i.e., from 0.3 wt % per day to 0.65 wt % per day).

On October 12, 1993, Millstone Unit No. 3 successfully conducted the second Type A test in the first 10-year service period. Test results indicated that the "As-Found" and "As-Left" ILRTs passed the technical specification acceptance criteria. The "As-Found" value was 0.1327 weight percent per day and the "As-Left" value was 0.1313 weight percent per day. These values represent 27.2% and 26.9% of the technical specification criterion of 0.4875 wt % per day (0.75 L_a), based on La equal to 0.65 wt % per day, respectively. In addition, as of October 9, 1993, the total Type B and C "As-Found" and "As-Left" leakage results were 0.099 wt % per day, and 0.084 wt % per day, respectively. These values represent approximately 25.3% and 21.5% of the technical specification limit of 0.39% wt % per day (0.6 L_a), based on L_a equal of 0.65 wt % per day, respectively. Correspondingly, the 1993 Type A, B, and C test results indicate that the "As-Found" and "As-Left" result in each test case was below the existing Technical Specification limit of 0.3 wt % per day. This further demonstrates the overall leakage integrity of the containment and its boundaries.

Based on the relatively low "As-Left" ILRT leakage rate (i.e., 0.1313 wt % per day is well below the existing technical specification limit of 0.225 wt % per day (0.75 La), based on

La equal to 0.3 wt % per day), which represents the overall containment integrated leakage rate for the containment prior to start-up, there is reasonable assurance that containment integrity will be maintained below the allowable leakage rate limit of 0.65 wt % per day. In addition, the total Type B and C "As-Left" leakage result of 0.084 wt % per day (this is well below the existing technical specification limit of 0.18 wt % per day (0.6 L_a), based on L_a equal to 0.3 wt % per day), provides further assurance that leakage, based on individual penetration, will be maintained within sufficient margin of the leakage limits.

Because the last Type A, B, and C tests were performed under the technical specification limit of 0.65 wt % per day, the proposed change to restore L_a to 0.65 wt % per day has no impact to these systems from a leakage allowance perspective. As indicated above, the previous test results met the technical specification leakage limits (based on 0.65 wt % per day) within sufficient margin and, therefore, would not present any challenge to these leakage limits.

NNECO has evaluated the proposed changes to Surveillance Requirement 4.6.6.1.d.3 that increase the time to draw a final required negative pressure as measured at the 24'-6" elevation of the auxiliary building in conjunction with the proposed change to reinstate the containment integrated leakage rate of 0.65 wt % per day to determine the impact on the offsite doses following a LOCA. The calculated radiological doses are, in most cases, less than the previously calculated doses (i.e., EAB and LPZ doses) and are within the 10CFR100 limits. Previously, the EAB thyroid and whole body doses as documented in the November 4, 1993, submittal were calculated to be 141 REM and 9.4 REM respectively, while the previously docketed (i.e., the November 4, 1993, submittal) LPZ doses to the thyroid and whole body were calculated to be 29.8 REM and 1.7 REM respectively. Utilizing the revised application of containment recirculation spray DF, the EAB thyroid and whole body doses were calculated to be 61 REM and 16.7 REM, respectively, and the LPZ thyroid and whole body doses were calculated to be 10.9 REM and 2.8 REM respectively. The assumptions used in the above radiological dose calculations are provided in Attachment 1. It is noted that a LOCA at Millstone Unit No. 3 is also one of the bounding accidents for the Millstone Unit No.3 control room, Millstone Unit No. 2 control room, and the Millstone Technical Support Center

habitability analysis. Therefore, the doses for these areas were recalculated and are presented in the Safety Assessment section above. The Millstone Unit No. 1 control room and the Emergency Operating Facility doses are bounded by the Millstone Unit No. 1 LOCA calculations.

The Millstone Unit Nos. 2 and 3 control rooms and Millstone Technical Support Center doses were not recalculated in 1993 (i.e., November 4, 1993, submittal) since EAB/LPZ doses proved that the releases were less than the 1990 submittal. In summary, all control room and Technical Support Center doses are within the guidelines of GDC 19. Therefore, the proposed changes do not result in an increase in consequences of an accident (i.e., a LOCA) previously analyzed.

The proposed changes to Bases Section 3/4.6.6 do not have any safety impact since they only reflect the changes proposed to Surveillance Requirement 4.6.6.1.d.3.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes do not compromise the ability of the SLCRS and ABFS to mitigate the consequences of an accident. The proposed changes do not make any physical or operational changes to existing plant structures, systems or components. The proposed changes do not introduce any new or unique operational modes or accident precursors. Therefore, the proposed changes do 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.

   NNECO has evaluated the proposed changes to Surveillance Requirement 4.6.6.1.d.3 that increase the time to draw a final required negative pressure as measured at the 24'-6" elevation of the auxiliary building in conjunction with the proposed change to reinstate the containment integrated leakage rate of 0.65 wt % per day to determine the impact on the offsite doses following a LOCA. The calculated radiological doses are, in most cases, less than the previously calculated doses and these doses are within the 10CFR100 limits. All control rooms and technical support center doses are within the guidelines of GDC 19. Therefore, the proposed changes do not involve a

significant reduction in the margin of safety.

Moreover, the commission 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 a SHC. The proposed changes to Sections 3.6.1.2.a and 4.6.6.1.d.3 of the Millstone Unit No. 3 Technical Specifications are not enveloped by a specific example. However, it has been demonstrated that the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

Environmental Considerations

NNECO has reviewed the proposed license amendment request against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve an SHC, do not significantly increase the types and amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, NNECO concludes that the proposed changes meet the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements of an environmental impact statement.

Nuclear Review Board Review

Millstone Unit No. 3 Nuclear Review Board has reviewed this proposed amendment and concurs with the above determination.

State Notification

In accordance with 10CFR50.91(b), we are providing the State of Connecticut with a copy of this proposed amendment to ensure their awareness of this request.

Schedule Required for NRC Approval

Currently, the next refueling outage is scheduled to begin in April 1995, with startup scheduled for June 1995. NNECO requests that this proposed license amendment be reviewed and approved prior to the start of the next cycle operation.

If the NRC Staff should have any questions or comments regarding this submittal, please contact Mr. R. G. Joshi at (203) 440-2080.

We will provide any additional information the NRC Staff may need to respond to this request, and we appreciate your efforts in support of this request.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Executive Vice President

cc: T. T. Martin, Region I Administrator
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin T.A. McCarthy, Director
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
P.O. Box 5066
Hartford, CT 06102-5066

Subscribed and sworn to before me

this 14th day of December, 1994

Lorraine J. D'Amico

Date Commission Expires: 3/31/98

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Supplementary Leak Collection and Release System
Radiological Dose Calculation Assumptions

December 1994

Attachment 1

Radiological Dose Calculation Assumptions

I. Offsite Doses

Figure 1 shows the assumed release pathways for the loss-of-coolant accident (LOCA) dose calculation. There are three release pathways analyzed: (1) containment leakage into the secondary containment which is filtered and released via the ventilation vent, (2) containment leakage that bypasses the secondary containment and is released unfiltered at the ground level, and (3) ESF leakage which is filtered and released from 0the ventilation vent. The radiological doses are calculated separately and then added. The assumption for each pathway are provided in Tables 1, 2 and 3 respectively. Table 4 presents the assumptions associated with the iodine removal from containment atmosphere by the containment spray systems. Table 8 provides the results of the dose calculations.

II. Millstone Unit No. 3 Control Room Dose

Table 5 presents the assumptions associated with the Millstone Unit No. 3 control room dose calculations. The CRADLE code was used to calculate the doses and the Millstone Unit No. 3 control room doses are presented in Table 8. The resulting doses are within the guidelines of General Design Criterion 0(GDC) 19.

III. Millstone Unit No. 2 Control Room Doses

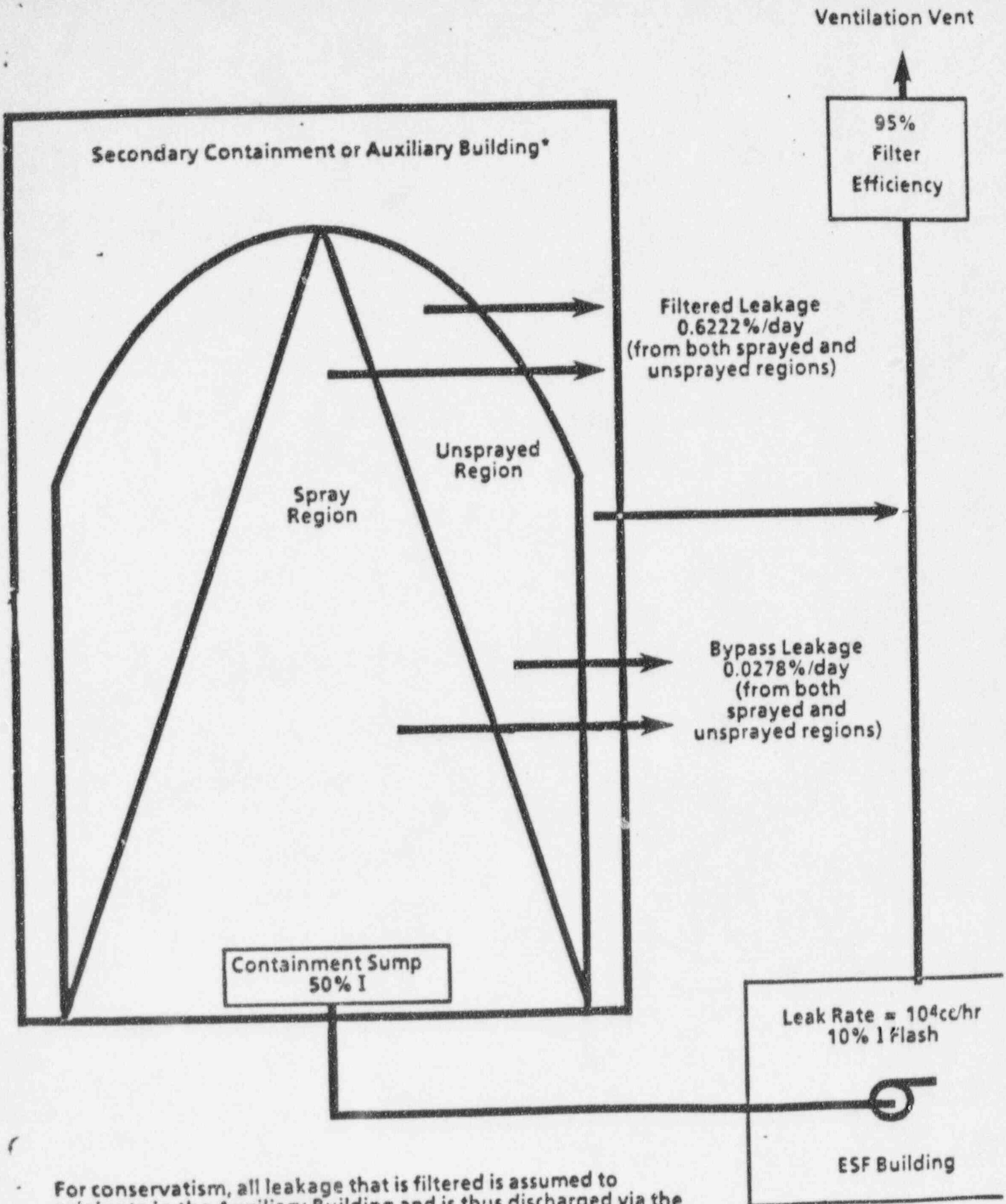
Table 6 presents the assumptions associated with the Millstone Unit No. 2 control room dose calculations from a Millstone Unit No. 3 LOCA. Using these assumptions and the CRADLE code resulted in the doses presented in Table 8. The resulting doses are within the guidelines of GDC 19.

IV. Millstone Technical Support Center

The effect of a Millstone Unit No. 3 LOCA on the dose to personnel in the Millstone Technical Support Center were also 0evaluated. The assumptions used for the dose calculations

are Opresented in Table 7. Using these assumptions and the CRADLE code resulted in the doses presented in Table 8. The Oresulting doses are within the guidelines of GDC 19.

Figure 1



* For conservatism, all leakage that is filtered is assumed to originate in the Auxiliary Building and is thus discharged via the ventilation vent. Leakage into the secondary containment would go to SLCRS and to Unit 1 stack. MP1 stack releases would result in lower doses.

TABLE 1

CONTAINMENT FILTERED LEAKAGE ASSUMPTIONS FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

	November 4, 1993 Submittal	Current Calculations
Power Level (Mwt)	3636	3636
Core inventory	TACT code output	FSAR Table 15.0-7 S&W code output
Iodine Chemical Form Elemental Particulate Organic	 95.5% 2.5% 2.0% (SRP 6.5.2 Rev. 1)	 91% 5% 4% (SRP 6.5.2 Rev. 2)
Offsite Breathing Rate (0-8) hrs (8-24) hrs (24-720) hrs	 3.47 E-4 m^3/sec 1.75 E-4 m^3/sec 2.32 E-4 m^3/sec	 3.47 E-4 m^3/sec 1.75 E-4 m^3/sec 2.32 E-4 m^3/sec
Release Rate	0.287% Containment Volume/Day	0.6222% Containment Volume/Day
Filter Efficiency All Form of Iodine	95%	95%
Release Rate After 24 hours	0.144% Containment Volume/Day	0.311% Containment Volume/Day
Release Point	Ventilation Vent	Ventilation Vent
Containment Free Air Volume	2.32 E+6 ft^3	2.32 E+6 ft^3
Site Boundary (EAB) x/Q's (sec/m^3)	4.3 E-4	4.3 E-4
Low Population Zone (LPZ) x/Q's (sec/m^3) (0-8) hrs (8-24) hrs (24-96) hrs (96-720) hrs	 2.91 E-5 1.99 E-5 8.66 E-5 2.63 E-6	 2.91 E-5 1.99 E-5 8.66 E-6 2.63 E-6
Time for the Secondary Containment to Achieve a Negative Pressure	2 minutes (2 minutes of total unfiltered leakage assumed)	<1 min. for -0.1 in water gauge <3 min. for -0.4 in water gauge (no unfiltered leakage assumed)
Thyroid Dose Conversion Factors	R.G. 1.109	ICRP30

TABLE 2
CONTAINMENT BYPASS LEAKAGE ASSUMPTIONS FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

	November 4, 1993 Submittal	Current Calculations
Power Level (Mwt)	3636	3636
Core Inventory	TACT Code Output	FSAR Table 15.0-7 S&W Code Output
Iodine Chemical Form Elemental Particulate Organic	 95.5% 2.5% 2% (SRP 6.5.2 Rev. 1)	 91% 5% 4% (SRP 6.5.2 Rev. 2)
Offsite Breathing Rate (0-8) hrs (8-24) hrs (24-720) hrs	 3.47 E-4 m^3/sec 1.75 E-4 m^3/sec 2.32 E-4 m^3/sec	 3.47 E-4 m^3/sec 1.75 E-4 m^3/sec 2.32 E-4 m^3/sec
Contain Leakage Rate L_a % Volume/Day	0.3%	0.65%
Bypass Leakage Rate % Volume/Day 0-24 hrs 24 hrs - 30 days	 0.01283% 0.006416%	 0.0278% 0.0139%
Release Point	Containment (Ground)	Containment (Ground)
Containment Free Volume (ft^3)	2.32 E6	2.32 E6
EAB X/Q's (sec/m^3)	5.42 E-4	5.42 E-4
LPZ X/Q's (sec/m^3) (0-8) hrs (8-24) hrs (24-96) hrs (96-720) hrs	 2.91 E-5 1.99 E-5 8.66 E-6 2.63 E-6	 2.91 E-5 1.99 E-5 8.66 E-6 2.63 E-6
Thyroid Dose Conversion Factor	R.G. 1.109	ICRP 30

TABLE 3
ESF LEAKAGE ASSUMPTIONS FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

	November 4, 1993 Submittal	Current Calculations
Power Level (Mwt)	3636	3636
Core Inventory	TACT Code Output	FSAR Table 15.0-7 S&W Code Output
Iodine Chemical Form Elemental Particulate Organic	 95.5% 2.5% 2.0% (SRP 6.5.2 Rev. 1)	 91% 5% 4% (SRP 6.5.1 Rev. 2)
Offsite Breathing Rate 0-8) hrs (8-24) hrs (24-720)	 3.47 E-4 1.75 E-4 2.32 E-4	 3.47 E-4 1.75 E-4 2.32 E-4
Containment Sump Volume (gallons) 220 sec - 1 hr 1 hr - 2 hrs >2 hrs	 80,000 700,000 1,000,000	 80,000 700,000 1,000,000
Iodine Released from Sump Water	10%	10%
Release Point	Ventilation Vent	Ventilation Vent
ESF Leakage - Twice the Maximum Operational Leakage	10,000 cc/hr	10,000 cc/hr
ESF Leakage Begins	220 seconds	220 seconds
EAB X/Q's (sec/m^3)	4.3 E-4	4.3 E-4
LPZ X/Q's (sec/m^3) (0-8) hr (8-24) hr (24-96) hr (96-720) hr	 2.91 E-5 1.99 E-5 8.66 E-6 2.63 E-6	 2.91 E-5 1.99 E-5 8.66 E-5 2.63 E-6
Thyroid Dose Conversion Factor	R.G. 1.109	ICRP 30

Table 4
Containment Spray Assumptions FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

	November 4, 1993 Submittal	Current Calculations
Assumed Quench Spray Initiation	0 Seconds	64 seconds
Recirculation Spray Initiation	750 Seconds	750 Seconds
Maximum Allowed DF from Spray Operation	12 Elemental	200 Elemental
Time to Achieve MAX Elemental Iodine DF	1.0 hrs	2.61 hrs
Containment Free Volume	2.32 E6 ft^3	2.32 E6 ft^3
Node 1 Sprayed Region Volume	1.206 E+06 ft^3	1.166 E+06 ft^3
Node 2 Unsprayed Region Volume	1.114 E+06 ft^3	1.154 E+06 ft^3
Mixing Rate = 2 turnovers/hr Unsprayed Region	2.227 E+06 ft^3/hr	2.308 E+06 ft^3/hr
Elemental Iodine Removal Coefficient λ spray λ plate out	28.1/hr 0.176/hr	20.0/hr 3.1/hr
Particulate Iodine Removal Coefficient λ DF <50 λ DF >50	2.16/hr 2.16/hr	12.5/hr 1.3/hr
Time to Achieve Particulate Iodine Depletion by a Factor of 50	N/A	2.07 hrs

Table 5
Millstone Unit No.3 (MP3) Control Room Parameters/Assumptions
FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

ASSUMPTIONS FOR CURRENT CALCULATIONS*	
(1)	Control room (CR) damper closure time = 3 sec.
(2)	Control room is pressurized from bottled air instantaneously 1 minute following control building isolation signal and bottle lasts 1 hour.
(3)	Control room intake prior to pressurization (<1 min) = 125 cfm
(4)	Minimum distance between CR intake and containment = 72 meters
(5)	Minimum distance between CR intake and ventilation vent = 38 meters
(6)	Wind velocity used in MP3 containment X/Q analysis = 1.9 m/sec
(7)	Wind velocity used in MP3 ventilation vent X/Q analysis = 1.7 m/sec
(8)	Time for plume radiation to reach intake from containment = 1.053E-2 hr
(9)	Time for plume radiation to reach intake from ventilation vent = 6.194E-3 hr
(10)	Unfiltered inleakage after 1.01667 hr when bottled air is exhausted and filtered intake begins = 10 cfm
(11)	Control room emergency ventilation rate after 1 hr: Filtered intake = 250 cfm Filtered recir = 750 cfm
(12)	Control room iodine cleanup rate = 0.1796/hr
(13)	Control room filter efficiency = 95% for all forms of iodine
(14)	Elemental iodine removal rate in containment (.01778 - 2.61) hr = 2.038/hr
(15)	Control Room Volume = 2.38E5 ft^3
(16)	Control Room X/Q's (sec / m^3)": Containment / Vent (0-8) hr: 8.08E-4 / 2.24E-3 (8-24) hr: 5.49E-4 / 1.40E-3 (24-96) hr: 1.95E-4 / 5.08E-4 (96-720) hr: 2.75E-5 / 9.68E-5
(17)	Particulate iodine removal rate in containment (.0118-2.07) hr = 1.90/hr (2.08 - 8) hr = 0.534/hr
(18)	Bypass release rate = 1.158E-5/hr

Table 5
Millstone Unit No. 3 (MP3) Control Room Parameters/Assumptions
FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

ASSUMPTIONS FOR CURRENT CALCULATIONS*
(19) Filtered release rate = 2.593E-4/hr
(20) Bypass release rate after 1 hr is one half = 5.792E-6/hr
(21) Filtered release rate after 1 hr is one half = 1.296E-4/hr
(22) Core Inventory from FSAR 15.0-7
(23) Thyroid Dose Conversion Factors from ICRP 30

*The Control Rooms and Technical Support Center dose calculations were not performed in 1993 since the EAB/LPZ doses proved that the releases were less than the February 26, 1990 submittal.

Table 6

Millstone Unit No. 2 (MP2) Control Room Parameters/Assumptions
FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

ASSUMPTIONS FOR CURRENT CALCULATIONS*	
(1)	Control Room Volume = 6.6E4 ft^3
(2)	Maximum outside supply fan flow rate prior to isolation = 800 cfm
(3)	Recirculation flow rate through charcoal filters = 2500 cfm
(4)	Time at which recirculation starts through filters = 10 min
(5)	Unfiltered inleakage rate = 130 cfm
(6)	Filter efficiency = 90% (all forms of iodine)
(7)	Time for Control Room to isolate = 23.1 sec (18.1 sec for monitor response and signal plus 5 seconds for damper to close)

*The Control Rooms and Technical Specification Support Center dose calculations were not performed in 1993 since the EAB/LPZ doses proved that the releases were less than the February 26, 1990 submittal.

Table 7
Millstone Technical Support Center (TSCC) Parameters/Assumptions
FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

ASSUMPTIONS FOR CURRENT CALCULATION*	
(1)	TSC damper closure time/isolation = 3.7 sec
(2)	TSC is isolated for the first 30 minutes
(3)	Wind velocity used in MP3 containment X/Q analysis = 1.9 m/sec
(4)	Wind velocity used in MP3 ventilation vent X/Q analysis = 1.9 m/sec
(5)	Time for plume radiation to reach intake from containment = 1.05E-2 hr
(6)	Time for plume radiation to reach intake from ventilation vent = 5.555E-3 hr
(7)	TSC unfiltered intake prior to isolation = 100 cfm
(8)	TSC unfiltered inleakage during the first 30 minutes = 50 cfm
(9)	No inleakage after pressurized at 30 minutes
(10)	TSC emergency ventilation rate: (0-30) minutes: Filtered intake = 0 cfm Filtered recir = 2000 cfm (> 30) minutes: Filtered intake = 100 cfm Filtered recir = 1900 cfm
(11)	TSC iodine cleanup rate: (0-30) minutes: = 3.434/hr (> 30) minutes: = 3.262/hr
(12)	TSC filter efficiency = 95% for all forms of iodine
(13)	Elemental iodine removal rate in containment (.01788 - 2.61) hr = 2.038 hr
(14)	TSC Volume = 3.32E4 ft^3
(15)	Occupancy Factors: (0-8) hr = 1.0 (8-24) hr = 0.5 (24-96) hr = 0.6 (96-720) hr = 0.4
(16)	TSC X/Qs (sec / m^3):

	Containment	Vent
(0-8) hr	8.08E-4	2.00E-3
(8-24) hr	2.69E-4	6.65E-4
(24-96) hr	1.92E-4	4.75E-4
(96-720) hr	3.01E-5	7.45E-5

Table 7

Millstone Technical Support Center (TSC) Parameters/Assumptions
FOR DOSE DUE TO THE MILLSTONE UNIT NO. 3 LOCA

ASSUMPTIONS FOR CURRENT CALCULATION*
(17) Particulate iodine removal rate in containment (.01778-2.07) hr = 1.90/hr (2.07 - 8) hr = 0.534/hr
(18) Bypass release rate = 1.158E-5/hr
(19) Filter release rate = 2.593E-4/hr
(20) Bypass release rate after 1 hr is one half = 5.792E-6/hr
(21) Filtered release rate after 1 hr is one half = 1.296E-4/hr
(22) Core Inventory from FSAR 15.0-7
(23) Thyroid Dose Conversion Factors from ICRP 30

*The Control Rooms and Technical Specification Support Center dose calculations were not performed in 1993 since the EAB/LPZ doses proved that the releases were less than the February 26, 1990 submittal.

Table 8
Dose Calculation Results

TYPE OF DOSE	LIMIT	FEB. 26, 1990 SUBMITTAL	NOV. 4, 1993* SUBMITTAL	CURRENT
EAB - Thyroid	300 REM	150 REM	141 REM	61 REM
EB - Whole Body	25 REM	19.5 REM	9.4 REM	16.7 REM
LPZ - Thyroid	300 REM	31.6 REM	29.8 REM	10.9 REM
LPZ - Whole Body	25 REM	3.5 REM	1.7 REM	2.8 REM
MP3 Control Room - Thyroid	30 REM	26 REM	-	7.9 REM
MP3 Control Room - Whole Body	5 REM	3.05 REM	-	4.1 REM
MP3 Control Room - Skin	30 REM	24.5 REM	-	25.5 REM
MP2 Control Room - Thyroid	30 REM	18.4 REM	-	10.3 REM
MP2 Control Room - Whole Body	5 REM	0.5 REM	-	2.4** REM
MP2 Control Room - Skin	30 REM	8.3 REM	-	8.2 REM
Tech Support Center - Thryoid	30 REM	7.4 REM	-	[illegible].3 REM
Tech Support Center - Whole Body	5 REM	1.4 REM	-	2.3 REM
Tech Support Center - Skin	30 REM	24.9 REM	-	29.90 REM

*The Control Room and Technical Support Center dose calculations were not performed in 1993 since the EAB/LPZ doses proved that the releases were less than the February 26, 1990 submittal.

**The previous calculations did not include shine dose from sources outside control room. For comparison purposes, the dose from airborne activity inside the control room is 0.5 REM, which is the same as the 1990 submittal.