Sanders, Carleen

From:	Nelson, Robert
Sent:	Monday, August 03, 2009 1:38 PM
To:	King, Mike
Cc:	Sanders, Carleen
Subject:	Action: Placement of Document in ADAMS

The letter from David Jaffe, NRC, to Edward J. Mroczka, Connecticut Yankee Atomic Power Co., dated January 25, 1991, Docket 50-423, "Issuance of Amendment (TAC No. 76066)" is considered to be the original and should be entered in ADAMS.

Carleen Sanders of our staff will deliver the document to you shortly.

Thanks,

R.A. Nelson

Robert A. Nelson
Deputy Director
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"Designated Original"

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 25, 1991

Docket No. 50-423

RECEIVED

JAN 28 1991

SENIOR VICE PRESIDENT
Nuclear Engineering & Operations

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 76066)

The Commission has issued the enclosed Amendment No. 59 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit No. 3, in response to your application dated February 26, 1990, as supplemented April 30, December 6 and 19, 1990.

The amendment modifies the Technical Specifications to allow an increase in the normal containment pressure range. The revised containment pressure range is 10.6 psia to 14.0 psia.

A copy of the related Safety Evaluation is also enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 59 to NPF-49
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 59
License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated February 26, 1990, as supplemented April 30, December 6 and 19, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 59 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 25, 1991

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ATTACHMENT TO LICENSE AMENDMENT NO. 59

FACILTIY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
viii	viii
ix	ix
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-7	3/4 6-7
3/4 6-8	3/4 6-8
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves or operator action during periods when containment isolation valves are opened under administrative control,** and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 53.27 psia (38.57 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

** The following manual valves may be opened on an intermittent basis under administrative control. 3FPW-V661, 3FPW-666, 3SSP-V13, 3SSP-V14, 3HCS-V2, 3HCS-V3, 3HCS-V9, 3HCS-V10, 3HCS-V6, 3HCS-V13, 3SAS-V875, 3SAS-V50, 3CHS-V371, 3CCP-V886, 3CCP-V887, 3CVS-V13.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.65% by weight of the containment air per 24 hours at P_a, 53.27 psia (38.57 psig);

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a; and

c. A combined leakage rate of less than or equal to 0.042 L_a for all penetrations identified in Table 3.6-1 as Enclosure Building bypass leakage paths when pressurized to P_a.

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

ACTION:

With the measured overall integrated containment leakage rate exceeding 0.75 L_a, or the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, or the combined bypass leakage rate exceeding 0.042 L_a, restore the overall integrated leakage rate to less than 0.75 L_a, the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.042 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using methods and provisions of ANSI N45.4-1972 (Total Time Method) and/or ANSI/ANS 56.8-1981 (Mass Point Method):

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 53.27 psia (38.57 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

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c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the test by verifying that the supplemental test results, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a, 53.27 psia (38.57 psig), at intervals no greater than 24 months except for tests involving:

1) Air locks

e. The combined bypass leakage rate shall be determined to be less than or equal to 0.042 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 53.27 psig (38.57 psig), during each Type A test;

f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

g. Purge supply and exhaust isolation valves shall be demonstrated OPERABLE by the requirements of Specifications 4.6.3.2.c and 4.9.9.

h. The provisions of Specification 4.0.2 are not applicable.

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TABLE 3.6-1

ENCLOSURE BUILDING BYPASS LEAKAGE PATHS

PENETRATION	DESCRIPTION	RELEASE LOCATION
14	N_2 to Safety Injection Tanks	Ground Release
15	Primary Water to Pressurizer Relief Tanks	Ground Release
35	Vacuum Pump Suction	Plant Vent
36	Vacuum Pump Suction	Plant Vent
37	Air Ejector Suction	Plant Vent
38	Chilled Water Supply	Plant Vent
45	Chilled Water Return	Plant Vent
52	Service Air	Turbine Building Roof Exhaust
54	Instrument Air	Turbine Building Roof Exhaust
56	Fire Protection	Ground Release
59	Fuel Pool Purification	Ground Release
60	Fuel Pool Purification	Ground Release
70	Demineralized Water	Ground Release
72	Chilled Water Supply	Plant Vent
85	Containment Purge	Ground Release
86	Containment Purge	Plant Vent
116	Chilled Water Return	Plant Vent
124	Nitrogen to Containment	Plant Vent

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 53.27 psia (38.57 psig).

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. 1) Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to P_a, 53.27 psia (38.57 psig), for at least 15 minutes;

or

2) Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to P_a, 53.27 psia (38.57 psig);

or

3) Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by completing an overall air lock leakage test per 4.6.1.3.b.

b. By conducting overall air lock leakage tests at not less than P_a, 53.27 psia (38.57 psig), and verifying the overall air lock leakage rate is within its limit:

1) At least once per 6 months,* and

2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

---

*The provisions of Specification 4.0.2 are not applicable.
**This represents an exemption to Appendix J, paragraph III.D.2.(b)(ii), of 10 CFR Part 50.

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CONTAINMENT SYSTEMS

CONTAINMENT PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment pressure shall be maintained greater than or equal to 10.6 psia and less than or equal to 14.0 psia.

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

ACTION:

With the containment pressure less than 10.6 psia or greater than 14.0 psia, restore the containment pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment pressure shall be determined to be within the limits at least once per 12 hours.

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does not exceed the design pressure of 60 psia during LOCA conditions. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature. The limits on the pressure and average air temperature are consistent with the assumptions of the safety analysis. The minimum total containment pressure of 10.6 psia is determined by summing the minimum permissible air partial pressure of 8.9 psia and the maximum expected vapor pressure of 1.7 psia (occurring at the maximum permissible containment initial temperature of 120°F).

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 60 psia in the event of a LOCA. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be locked closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The Type C testing frequency required by 4.6.1.2d is acceptable, provided that the resilient seats of these valves are replaced every other refueling outage.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH SPRAY SYSTEM and RECIRCULATION SPRAY SYSTEM

The OPERABILITY of the Containment Spray Systems ensures that containment depressurization and iodine removal will occur in the event of a LOCA. The pressure reduction, iodine removal capabilities and resultant containment leakage are consistent with the assumptions used in the safety analyses.

3/4.6.2.3 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.0 and 7.35 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 59

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By application for license amendment dated February 26, 1990, as supplemented April 30, December 6 and 19, 1990, Northeast Nuclear Energy Company, et al. (the licensee), requested changes to Millstone Unit 3 Technical Specifications (TS) regarding normal containment operating pressure. The current TS require that the containment pressure be maintained subatmospheric and be greater than 8.9 psia but less than or equal to 12 psia during operation Modes 1 through 4. The licensee proposed to change the containment operating pressure and associated TS to a new range between 10.6 psia and 14.0 psia.

2.0 DISCUSSION

Millstone Unit 3 is a dual-containment plant. The containment is comprised of a primary containment structure and a secondary containment enclosure building and an associated supplementary leak collection and release system (SLCRS). Containment entries are required for inspecting unidentified reactor coolant system leakage, investigating boron precipitation, and plant start-up surveillances or inspections. The risk of injury to plant personnel performing such physical labor in the subatomospheric containment has been found significant due to crossing the pressure boundary and also due to oxygen deficiency. Personnel are required to wear self-contained respirator (Rexnord "Bio-Packs") to supply supplemental oxygen but the environment of low pressure and high temperature in the containment causes significant potential for personnel injury during containment entries. The licensee stated that 38 personnel medical incidents had occurred due to containment entries during the past 4 years since the plant was licensed. In addition, the use of Bio-Packs cause personnel working in the containment to become less efficient.

In order to allow containment entry with a minimal pressure change and eliminate the need to carry heavy, awkward supplemental oxygen units (Bio-Packs), the licensee proposes to increase the containment operating pressure. In support of the TS change, the licensee performed safety analyses to assess the impact on the accidents evaluated as the design basis, the potential for creation of a new unanalyzed event, and the impact on the margin of safety. The staff's evaluation of the licensee's submittals is described below.

3.0 EVALUATION

The current containment parameters and the licensee proposed changes are listed in Table 1. The licensee's revised safety analyses are based on the proposed parameters.

Table 1

Containment Parameter	Current	Proposed Change
Normal Operating Pressure	9.8 psia	14.0 psia
Design Pressure	45 psig	45 psig
Peak Pressure (Pa)	36.1 psig	38.57 psig
Containment Leak Rate (La)	2912.68 SCFH (0.9 wt% per day)	2206.33 SCFH (0.65 wt% per day)
Secondary Containment Bypass Leakage Fraction	0.01La (0.009 wt% per day)	0.042La (0.028 wt% per day)
Service Water Temperature	75°F	75°F

3.1 Containment Integrity Analysis

3.1.1 Containment Pressure and Temperature Responses

Two loss-of-coolant-accident (LOCA) cases for containment pressure/temperature responses were reanalyzed by the licensee using the same methods and computer models as described in Section 6.2.1 of the Final Safety Analysis Report (FSAR) except the initial containment pressure was increased to 14.2 psig. The licensee reanalyzed the hot leg double-ended rupture (DER) and the pump suction DER with failure of one engineering safety features (ESF) train. The limiting accident for peak containment pressure was found to be the hot leg DER at 38.57 psig which was below the containment design pressure of 45 psig. Since the staff has previously reviewed and approved the methodology and analytical model, the staff concludes that the licensee's LOCA analysis is acceptable.

The pump suction DER with failure of one ESF train was found to be the limiting accident for the long term containment pressure transient. The current analysis showed that the containment pressure depressurized to atmospheric pressure in 41.33 minutes after a LOCA and then the containment

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pressure returned to subatmopheric. The licensee recalculated this pressure transient and the result showed that the containment pressure remains above atmospheric pressure for the duration of the accident. The staff's review found that containment pressure remaining above atmospheric would cause continued leakage from the containment. This will be further discussed in Section 3.3 of this evaluation.

3.1.2 Main Steam Line Break Analysis

The licensee recalculated the containment pressure response for a main steam line break (MSLB) for full DER at hot standby (zero power). The peak containment pressure based on a new containment operating pressure of 14.2 psia was calculated to be 34.5 psig which was below the peak containment pressure following a LOCA. The staff concludes that the MSLB reanalysis has a minor effect on the containment pressure responses.

3.1.3 Subcompartment Pressurization Analysis

The initial atmospheric conditions within the subcompartment which can maximize the differential pressure across the walls are the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. Increasing initial pressure will increase air mass in the compartment and reduce pressure difference across the walls. Therefore, the staff concludes that the proposed change has no effect on current containment subcompartment analysis.

3.1.4 Combustible Gas Concentration

The increased containment operating pressure will result in lower hydrogen concentration in the containment because the rate of hydrogen generation is unchanged but the mass of air in the containment is increased. Therefore, the staff concludes that the proposed change has no effect on current evaluation of hydrogen control.

3.2 Safety Systems Evaluation

3.2.1 Quench Spray System/Containment Recirculation System

The Quench Spray System (QSS) and the Containment Recirculation System (CRS) had previously been reviewed and approved by the NRC staff for their containment pressure reduction and core cooling roles, respectively. The licensee now proposes to credit the QSS and CRS for removal of post-LOCA fission products inside containment.

The NRC staff has reviewed the QSS and CRS against the criteria of Standard Review Plan (SRP) 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System." In a letter dated December 6, 1990, the licensee addressed the criteria of SRP 6.5.2, Revision 2 regarding the QSS and CRS.

The staff concludes that the containment spray system as a fission product cleanup system is acceptable and meets the relevant requirements of General Design Criterion 41, "Containment Atmosphere Cleanup," General Design Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems," and General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems." This conclusion is based on the following.

The concept upon which the proposed system is based has been demonstrated to be effective for iodine absorption and retention under post-accident conditions. The proposed system design is an acceptable application of this concept. The system provides suitable redundancy in components and features such that its safety function can be accomplished assuming a single failure. The staff concludes that the system meets the requirements of General Design Criterion 41.

The proposed pre-operational tests, post-operational testing and surveillance, and proposed limiting conditions of operation for the spray system provide adequate assurance that the iodine scrubbing function of the containment spray system will meet or exceed the effectiveness assumed in the accident evaluation and, therefore, meets the requirements of General Design Criteria 42 and 43.

3.2.2 Containment Air Recirculation System

The containment air recirculation (CAR) system is not designed to operate post-LOCA and is automatically shut down by a containment depressurization actuation signal. Therefore, the proposed change has no effect on the consequences of a DBA due to the CAR system performance.

3.2.3 Containment Vacuum System

The containment vacuum system reduces the containment pressure from atmospheric to subatmospheric using a vacuum ejector. The proposed change will result in less frequent operation of the vacuum pump in order to maintain the new subatmospheric pressure. The system is not safety related. Therefore, the staff concludes that the proposed change has no effect on the consequences of a DBA due to the containment vacuum system performance.

3.2.4 Containment Pressure Monitors

At the present time, there are two narrow range containment pressure transmitters (3LMS&PT43A and B) that provide indication in the control room for a containment pressure range of 8.5 to 13.5 psia during normal operation. These transmitters and associated instrumentation/displays will be modified prior to implementing the proposed changes to the TS to achieve a range of 8.5 to 14.5 psi as indicated in the licensee's letter dated December 19, 1990. We find this commitment to be acceptable.

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3.3 Containment Leakage Evaluation

The current containment integrity analysis assumed that the containment pressure would drop to approximately 4 psig within 1 hour after a LOCA and then the containment would be maintained subatmospheric for 30 days. The current containment integrated leak rate was set at La, or 0.9% by weight of the containment air per day (0.9 wt%/day), for the first hour of a LOCA and zero leakage after the containment returned to subatmopheric. The proposed change in containment operating pressure will result in containment pressure remaining above atmospheric for the duration of the accident and, therefore, continued containment leakage is assumed.

To compensate for the increased time in leakage release, the licensee proposed to reduce the TS allowable leak rate from 0.9 wt%/day to 0.65 wt%/day for the first 24 hours and 0.325 wt%/day after 24 hours until 30 days. The licensee stated that the proposed limit of 0.65 wt%/day represents the maximum containment allowable leakage in compliance with 10 CFR Part 100 requirements. The licensee provided containment integrated leak rate test (CILRT) results for the second refueling outage. The as-left containment leakage rate was 0.2919 wt%/day or 641 SCFH. The current acceptable leakage for the CILRT is 0.75La(0.9), or 0.675 wt%/day, which corresponds to an allowable leakage rate of 1428 SCFH. The proposed containment leakage rate is 0.75La(0.65), or 0.488 wt%/day, which corresponds to an allowable leakage rate of 1076 SCFH. The staff finds that the proposed containment leakage rate is equivalent to 0.52La which is less than 0.75La required by Appendix J to 10 CFR Part 50. Furthermore, the CILRTs were performed at Pa of 39.4 psig which was higher than the proposed new test pressure of 38.6 psig. The CILRT result would be lower if the tests were performed with the new test pressure. Based on the licensee provided information, the staff concludes that the proposed containment leakage rate is conservative and acceptable.

The licensee proposed to increase the secondary containment bypass leakage rate from 0.01La to 0.042La or 0.009 wt%/day to 0.028 wt%/day. The licensee performed a containment radiological leakage analysis to provide the maximum value achievable for bypass leakage and found that the increased bypass leakage still meets the 10 CFR Part 100 dose limit. The staff concludes that the proposed bypass leakage rate is acceptable.

3.4 Electric Equipment Qualification for Service Conditions

The current electric equipment qualification (EEQ) was based on a normal containment pressure range of 9.5 to 14.7 psia. The proposed containment operation pressure 14.2 psia falls within this range, and therefore, will not impact current EEQ. The licensee stated that the proposed increase in containment pressure would result in some increase in the radiation consequences following a DBA, but would not impact the existing accident radiation qualification of EEQ equipment. The staff confirmed the results of the radiation qualification and found that the calculated maximum radiation level was lower than the electric equipment tested values by more than 10%. This provided an acceptable margin for the radiation qualification of EEQ equipment. Therefore, the staff concludes that the current EEQ is acceptable.

4.0 POST LOCA DOSE ASSESSMENT

The original and current radiological consequence analyses were based on the sub-atmospheric design which terminates all primary containment leakage within 1 hour. Consequently, the proposed change in the containment pressure in itself, without modifying any other requirements, would result in an increase in calculated offsite radiological consequences in an event of a LOCA.

Therefore, in order to compensate for the potential increase in the post-LOCA offsite doses, the licensee claimed full credit for the iodine removal capabilities of the containment chemical spray in accordance with SRP Section 6.5.2, Revision 1. The licensee stated that such credit is not claimed for the original and current LOCA analysis since the radiological consequences were acceptable without the spray. The staff found in the Millstone Unit 3 Safety Evaluation Report (NUREG-1031) dated July 1984 that the radiological consequences were also acceptable without the containment spray credit for iodine removal.

In addition, the licensee also proposed to change the allowable containment leak rates as follows:

Allowable Leak Rates (volume percent per day)
TS Sections 3.1.6.2 and 3.1.6.4)

Primary Containment Leak Rate (La)	0 to 1.0 (hours)	1 to 24 (hours)	24 to 720 (hours)
Current	0.9	0	0
Proposed	0.65	0.325	0.325
Bypass Leakage			
Current	0.009	0.009	0.009
Proposed	0.042	0.042	0.042

Using the above proposed leak rates with a full credit allowed for iodine removal by the containment spray and the assumptions and parameters in Table 15.2 of Millstone Unit 3 SER, the staff computed the offsite doses for the

Millstone 3 Exclusion Area (EAB) and Low Population Zone (LPZ) boundaries. The computed offsite doses are listed in Table 2, are within the acceptance criteria given in Section 15.7.5 of the SRP and the exposure guidelines of 10 CFR Part 100 and are therefore acceptable.

TABLE 2
POST-LOCA OFFSITE DOSES
(rem)

	Original(1)	Revised(2)	Limit(3)
Exclusion Area Boundary			
Thyroid	158	265	300
Whole Body	21	24	25
Low Population Zone			
Thyroid	8	180	300
Whole Body	1.1	5.6	25

(1) Table 15.1 of Millstone 3 SER dated July 1984
(2) Staff recalculated values
(3) 10 CFR Part 100

5.0 PROPOSED CHANGES TO THE TS

The licensee has proposed the following changes to the TS:

1. The peak calculated containment pressure (P_a) would be changed to 53.27 psia (38.57 psig) in Sections 4.6.1.1.c, 3.6.1.2.a, 4.6.1.2.a, 4.6.1.2.d, 4.6.1.2.e, 3.6.1.3.b, 4.6.1.3.a.1 and a.2, 4.6.1.3.b.

2. The integrated leak rate at P_a, containment leak rate (L_a) would be changed from 0.9 weight percent per day to 0.65 weight percent per day in Section 3.6.1.2.a.

3. The combined bypass leakage rate would be changed from 0.01 L_a to 0.042 L_a in Sections 3.6.1.2 and 4.6.1.2.e.

4. The operating containment pressure of 14.0 psia would be specified in Section 3.6.1.4. In addition, the maximum and minimum limit for the containment pressure would be specified as total containment pressure instead of air partial pressure.

5. Figure 3.6.1 would be deleted as the containment pressure will be read directly from the main control board indicators.

6. TS Table 3.6-1 would be changed as follows:

   a. Penetrations Z-28 and Z-29 (aerated drains and gaseous vents) would be deleted.

   b. Penetrations Z-59, Z-60, and Z-124 (fuel pool purification and nitrogen supply to containment) would be added.

   c. Table 3.6.1 would be revised to include description for each penetration.

The proposed changes to the TS associated with the operating containment pressure and the associated peak calculated containment pressure (Pa), containment leak rate (La) and bypass leakage rates are supported by the analysis presented in Section 3, herein. The results of the analyses indicated that the potential post-LOCA off-site radiological consequences are within the limits of 10 CFR Part 100. Accordingly, the proposed changes to the TS are acceptable.

With regard to TS Table 3.6-1, "Enclosure Building Bypass Leakage Paths," the licensee has performed a review of the penetrations specified in this table whose combined leakage must be less than .01 La per TS 3.6.1.2. The licensee has determined that two penetrations, Nos. 28 and 29, do not represent potential leakage paths. Since potential leakage would occur within the Auxiliary Buildings, for these penetrations, the liquid would be maintained within the building while gaseous releases would be processed by the safety-grade ventilation systems. Accordingly, penetrations 28 and 29 should be deleted from TS Table 3.6-1. Conversely, the licensee has identified three penetrations, Nos. 59, 60 and 124, whose leakage could bypass the Enclosure Building and thus are appropriately added to TS Table 3.6-1. Finally, adding the proposed penetration descriptions to TS Table 3.6-1 does not effect either the associated Limiting Conditions for Operation or the Surveillance Requirements and is, thus, acceptable.

6.0 ENVIRONMENTAL CONSIDERATIONS

Pursuant to 10 CFR 51.21 and 51.35, an environmental assessment and finding of no significant impact was prepared and published in the Federal Register on December 20, 1990 (55 FR 52228). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 25, 1991

Principal Contributors:

J. Guo
D. Jaffe
J. Lee

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-423

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 59 to Facility Operating License No. NPF-49 issued to Northeast Nuclear Energy Company, which revised the Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 3 located in New London County, Connecticut. The amendment is effective as of the date of issuance.

The amendment modified the Technical Specifications to allow an increase in the normal containment pressure range. The revised containment pressure range is 10.6 psia to 14.0 psia.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on April 16, 1990 (55 FR 14149). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action, see (1) the application for amendment dated February 26, 1990, as supplemented April 30, December 6 and 19, 1990, (2) Amendment No. 59 to License No. NPF-49, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street N.W., Washington, D.C. and at the Learning Resources Center, Thames Valley State Technical College, 574 New London Turnpike, Norwich, Connecticut 06360. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - I/II.

Dated at Rockville, Maryland this 25th day of January 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation