March 10, 1999

Mr. Martin L. Bowling, J
Recovery Officer - Technical Services
Northeast Nuclear Energy Company
c/o Ms. Patricia A. Loftus
Director - Regulatory Affairs
P. O. Box 128
Waterford, Connecticut 06385

SUBJECT: ISSUANCE OF AMENDMENT - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 (TAC NOS. MA3410 AND MA3672)

Dear Mr. Bowling:

The Commission has issued the enclosed Amendment No. 228 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated August 12, 1998, as supplemented by letter dated October 30, 1998, and application dated September 28, 1998, as supplemented by letters dated January 7 and 20, 1999.

The amendment allows implementation of a revised main steamline break analysis and revised radiological consequences.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
/S/
Stephen Dembek, Project Manager
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 228 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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2. Safety Evaluation

cc w/encls: See next page

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Unit 2

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Millstone Nuclear Power Station
Unit 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 228
License No. DPR-65

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

   A. The applications for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated August 12, 1998, as supplemented by letter dated October 30, 1998, and September 28, 1998, as supplemented by letters dated January 7 and 20, 1999, comply 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 applications, 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, changes to the Final Safety Analysis Report are authorized to reflect the revised main steamline break analysis and radiological consequences analyses as set forth in the licensee's application dated September 28, 1998, as supplemented by letter dated January 20, 1999. Additionally, 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. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 228, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: March 10, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 228

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Remove	Insert
3/4 3-11	3/4 3-11
3/4 4-9	3/4 4-9
3/4 4-13	3/4 4-13
3/4 4-15	3/4 4-15
3/4 6-12	3/4 6-12
3/4 6-26	3/4 6-26
3/4 7-16	3/4 7-16
3/4 7-17	3/4 7-17
3/4 7-17a	3/4 7-17a
3/4 7-18	3/4 7-18
3/4 9-17	3/4 9-17
B 3/4 4-3	B 3/4 4-3
B 3/4 6-3	B 3/4 6-3
B 3/4 7-4	B 3/4 7-4
6-18	6-18
6-18a	6-18a
6-19	6-19

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

4.3.2.1.4 The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6}$ sec/m^3 shall be used when the system is aligned to purge through the building vent and a $X/Q = 7.5 \times 10^{-8}$ sec/m^3 shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and particulate (Half Lives greater than 8 days) radioactivity shall be assumed to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than 5 $\times 10^5$ cpm.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 0.035 GPM primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE shall be demonstrated to be within limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram of gross specific activity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours. Specification 3.0.4 is not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram of gross specific activity, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram of gross specific activity, perform the sampling and analysis requirements of item 4 a) of Table 4.4-2 until the specific activity of the primary coolant is restored to within its limits.

*With T_{avg} $\geq$ 515°F.

TABLE 4.4-2

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days
3. Radiochemical Analysis for $\bar{E}$ Determination	1 per 6 months*
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131, or 100/$\bar{E}$ μCi/gram of gross specific activity, and b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer. The provisions of Specification 4.0.4 are not applicable.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains, with each cooling train consisting of two containment air recirculation and cooling units, shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

Inoperable Equipment	Required Action
a. One containment spray train	a.1 Restore the inoperable containment spray train to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.
b. One containment cooling train	b.1 Restore the inoperable containment cooling train to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.
c. One containment spray train AND One containment cooling train	c.1 Restore the inoperable containment spray train or the inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
d. Two containment cooling trains	d.1 Restore at least one inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
e. All other combinations	e.1 Enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.6.2.1.1 Each containment spray train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each spray pump from the control room,

2. Verifying, that on recirculation flow, each spray pump develops a discharge pressure of $\geq$ 254 psig,

*The Containment Spray System is not required to be OPERABLE in MODE 3 if pressurizer pressure is < 1750 psia.

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the train starts on an Enclosure Building Filtration Actuation Signal (EBFAS).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent Control Room Emergency Ventilation Trains shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

Modes 1, 2, 3, and 4:

With one Control Room Emergency Ventilation Train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6*

a. With one Control Room Emergency Ventilation Train inoperable, restore the inoperable train to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation Train in the recirculation mode.

b. With both Control Room Emergency Ventilation Trains inoperable, or with the OPERABLE Control Room Emergency Ventilation Train required to be in the recirculation mode by ACTION (a.) not capable of being powered by an OPERABLE normal and emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

* In Modes 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then Limiting Condition for Operation (LCO) 3.7.6.1.a or 3.7.6.1.b shall be invoked as applicable.

4.7.6.1 Each Control Room Emergency Ventilation Train shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is $\leq$ 100°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room, flow through the HEPA filters and charcoal absorber train and verifying that the train operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

   1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 2500 cfm $\pm$ 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.* The carbon sample shall have a removal efficiency of $\geq$ 95 percent.

   3. Verifying a train flow rate of 2500 cfm $\pm$ 10% during train operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.4 inches Water Gauge while operating the train at a flow rate of 2500 cfm $\pm$ 10%.

2. Verifying that on a recirculation signal, with the Control Room Emergency Ventilation Train operating in the normal mode and the smoke purge mode, the train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that control room air in-leakage is less than 130 SCFM with the Control Room Emergency Ventilation System operating in the recirculation/filtration mode.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that on a Spent Fuel Storage Pool Area high radiation signal, the train automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

---

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The steam generator tube leakage limit of 0.035 GPM per steam generator ensures that the dosage contribution from the tube leakage will be less than the limits of General Design Criteria 19 of 10CFR50 Appendix A in the event of either a steam generator tube rupture or steam line break. The 0.035 GPM limit is consistent with the assumptions used in the analysis of these accidents.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses. The leak rate surveillance requirements assure that the leakage assumed for the system outside containment during the recirculation phase will not be exceeded.

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray system during post-LOCA conditions.

To be OPERABLE, the two trains of the containment spray system shall be capable of taking a suction from the refueling water storage tank on a containment spray actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal. Each containment spray train flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

The containment cooling system consists of two containment cooling trains. Each containment cooling train has two containment air recirculation and cooling units. For the purpose of applying the appropriate action statement, the loss of a single containment air recirculation and cooling unit will make the respective containment cooling train inoperable.

Either the containment spray system or the containment cooling system has sufficient heat removal capability to handle any design basis accident. However, the containment spray system is more effective in dealing with the superheated steam from a main steam break inside containment. In addition, the containment spray system provides a mechanism for removing iodine from the containment atmosphere. Therefore, at least one train of containment spray is required to be OPERABLE when pressurizer pressure is ≥ 1750 psia, and the allowed outage time for one train of containment spray reflects the dual function of containment spray for heat removal and iodine removal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The Technical Requirements Manual contains the list of containment isolation valves (except the containment air lock and equipment hatch). Any changes to this list will be reviewed under 10CFR50.59 and approved by the Plant Operations Review Committee (PORC).

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 FLOOD LEVEL

The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure operator accessibility to the intake structure action to provide pump motor protection will be initiated when the water level reaches plant grade level.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm $\pm$ 10%.

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), operator action will be required to return to a full train lineup. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train lineup. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until 10 minutes into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

d. Documentation of all failures (inability to lift or reclose within the tolerances allowed by the design basis) and challenges to the pressurizer PORVs or safety valves.

ANNUAL RADIOACTIVE EFFLUENT REPORT

6.9.1.6 A routine Annual Radioactive Effluent Report covering the operation of the unit during the previous calendar year of operation shall be submitted by May 1 of each year.

The report shall include that information delineated in the REMODCM.

Any changes to the REMODCM shall be submitted in the Annual Radioactive Effluent Report.

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.8 a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

3/4.1.1.1	SHUTDOWN MARGIN - T_{avg} > 200°F
3/4.1.1.2	SHUTDOWN MARGIN - T_{avg} ≤ 200°F
3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.6	Regulating CEA Insertion Limits
3/4.2.1	Linear Heat Rate
3/4.2.3	Total Integrated Radial Peaking Factor - F_r^T
3/4.2.6	DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) XN-75-27(A) and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P).

2) ANF-84-73 Revision 5 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels, July 1990.

CORE OPERATING LIMITS REPORT (CONT.)

3) XN-NF-82-21(P)(A) Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, September 1983.

4) EMF-84-93(P) Revision 1, "Steamline Break Methodology for PWRs," Siemens Power Corporation, June 1998.

5) XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983.

6) XN-NF-82-49(P)(A) Revision 1, "EXXON Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, April 1989.

7) XN-NF-82-49(P)(A) Revision 1 Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation, December 1994.

8) EXEM PWR Large Break LOCA Evaluation Model as defined by:

- XN-NF-82-20(P)(A) Revision 1 Supplement 2, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, February 1985.
- XN-NF-82-20(P)(A) Revision 1 and Supplement 1, 3, and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Advanced Nuclear Fuels Corporation January 1990.
- XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982.
- XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.
- ANF-81-58(P)(A) Revision 2 Supplements 3 and 4, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," Advanced Nuclear Fuels Corporation, June 1990.
- XN-NF-85-16(P)(A) Volume 1 and Supplements 1, 2, and 3; Volume 2, Revision 1 and Supplement 1, "PWR 17 x 17 Fuel Cooling Test Program," Advanced Nuclear Fuels Corporation, February 1990.
- XN-NF-85-105(P)(A) and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Advanced Nuclear Fuels Corporation, January 1990.

CORE OPERATING LIMITS REPORT (CONT.)

9) XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized water reactors," Exxon Nuclear Company, October 1983.

10) XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, September 1983.

11) XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.

12) ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991.

13) XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1988.

14) ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992.

15) XN-NF-507(P)(A) Supplements 1 and 2, "ENC Setpoint Methodology for C.E. Reactors: Statistical Setpoint Methodology," Exxon Nuclear Company, September 1986.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 228

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated August 12, 1998, as supplemented by letter dated October 30, 1998, the Northeast Nuclear Energy Company, et al. (NNECO, or the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 2 Technical Specifications (TS) regarding the Main Steamline Break (MSLB) Analysis. Additionally, by letter dated September 28, 1998, as supplemented by letters dated January 7 and 20, 1999, the licensee submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 2 TS regarding the Control Room Ventilation System and changes to the Final Safety Analysis Report (FSAR) regarding revised radiological consequences analyses. The staff determined that these two amendment requests would be reviewed and approved in one license amendment. The supplemental submittals provided additional information that did not change the staff's proposed no significant hazards consideration determinations.

2.0 BACKGROUND

In early 1998, during an engineering review of the MSLB analysis presented in FSAR Section 14.1.5, NNECO found (Licensee Event Report (LER)-98-007-00 dated April 8, 1998) that non-conservative assumptions related to the power distributions and reactivity data were contained in the calculation, which supports the existing MSLB analysis. The nonconservative assumptions may result in violation of the safety limits of the fuel design. By letters of August 12, 1998, and September 28, 1998, the licensee submitted the MSLB reanalysis to support operations of Millstone Unit 2 Cycle 13 and future cycles. The MSLB reanalysis was performed with the revised MSLB methodology (Ref. 1) developed by Siemens Power Corporation (SPC). To maintain updated TS, the licensee proposed changes to TS 6.9.1.8b to list the updated references that describe the methods (including the revised MSLB methodology) used by the licensee to perform the safety analysis.

The radiological consequences of the MSLB were evaluated by NNECO as part of a reanalysis of the MSLB break accident. The reanalysis projected limited fuel failure where the previous analysis projected none. In order to minimize the potential radiological consequences of the increased fuel failure, NNECO has proposed reducing the TS maximum allowable steam generator tube leakage from 1.0 gallon per minute (gpm) to 0.035 gpm per steam generator.

The loss-of-coolant accident (LOCA) analysis was revised by NNECO as part of its effort to upgrade design basis accident analyses. The analysis assumptions modified include: (1) credit for iodine removal by containment sprays; (2) core inventory based on extended burnup fuel and a higher power level than currently allowed by the operating license; (3) incorporation of refueling water storage tank (RWST) backleakage release source; (4) updated atmospheric dispersion factors; (5) reduced control room volume and recirculation flow; (5) increased control room in-leakage; and (6) revision in parameters associated with emergency core cooling system leakage.

The revised MSLB and LOCA analyses take credit for equipment not previously assumed in the analyses and for plant or equipment operating restrictions not currently addressed in the TS. This application proposes changes to several TS to address these revised analysis assumptions. The proposed TS changes are:

TS 3.3.2.1, *Instrumentation-Engineered Safety Features Actuation System,* would be revised to make editorial changes.

TS 3.4.6.2, *Reactor Coolant System-Reactor Coolant System Leakage*, would be revised to reduce the maximum allowable primary-to-secondary leakage to 0.035 gpm per steam generator. Supporting changes to leakage test requirements were also proposed.

TS 3.4.8, *Reactor Coolant System-Specific Activity*, would be revised to clarify language regarding specific activity limiting conditions for operation (LCOs) and surveillance testing.

TS 3.6.2.1, *Containment Systems-Depressurization and Cooling Systems Containment Spray and Cooling Systems*, would be revised to reduce the allowed outage time of one containment spray train from 7 days to 72 hours.

TS 3.6.5.1, *Containment Systems-Secondary Containment Enclosure Building Filtration System*, would be revised to reduce the maximum allowable pressure drop across the combined high-efficiency particulate air (HEPA) filters and charcoal absorber banks from 6 inches water gauge to 2.6 inches water gauge.

TS 3.7.6.1, *Plant Systems-Control Room Emergency Ventilation System*, would be revised to (1) reduce the maximum allowable pressure drop across the combined HEPA filters and charcoal absorber banks from 6 inches water gauge to 3.4 inches water gauge, (2) increase maximum allowable control room air in-leakage from 100 standard cubic feet per minute (scfm) to 130 scfm, and (3) clarify language related to initial conditions for testing ventilation switchover to recirculation mode.

TS 3.9.15, *Refueling Operations-Storage Pool Area Ventilation System-Fuel Storage*, would be revised to reduce the maximum allowable pressure drop across the combined HEPA filters and charcoal absorber banks from 6 inches water gauge to 2.6 inches water gauge.

TS 6.9.1.8, *Core Operating Limits Report*, would be revised to reference the new analytical methods used by the licensee.

2.0 EVALUATION

2.1 MSLB - Reactor Systems Analytical Methods

The licensee used the analytical methodology described in Topical Report EMF-84-093(P), Revision 1 (Ref. 1), to perform the MSLB analysis. The MSLB analysis involved three computer codes: ANF-RELAP for the reactor coolant system (RCS) response calculation; XTGPWR for the detailed core neutronics calculation and XCOBRA-IIIC for the detailed core thermal-hydraulic calculation. The ANF-RELAP results were used as input to the XTGPWR code which calculated core power distributions and reactivity. The XTGPWR calculations were coupled to the XCOBRA-IIIC calculations by transferring the XCOBRA-IIIC nodal moderator densities into XTGPWR and iterating between XTGPWR and XCOBRA-IIIC until the power distribution converged. With the converged power distribution thus calculated and necessary input from XTGPWR, the XCOBRA-IIIC calculations provided the subchannel analysis of margin to departure from nucleate boiling.

The generic acceptability of the MSLB methods is still under staff review. However, the staff's review has progressed to the stage that the staff has determined that the methods are acceptable for the MSLB analysis at Millstone Unit 2 because the staff has found that (1) the MSLB methods apply the previously approved computer codes (ANF-RELAP, XTGPWR, and XCOBRA-IIIC) and critical heat flux (CHF) correlations (the XNB and modified Barnett correlation) for calculations of the fuel and system responses, (2) the method for power distribution calculations using XTGPWR and XCOBRA-IIIC provides convergent and consistent results, and, thus, is acceptable, and (3) the licensee's applications of the MSLB methods are within the applicable ranges of the approved computer codes and CHF correlations for MSLB analyses.

2.1.1 Analytical Results

Following an MSLB event, the steam release increases at the beginning of the transient and decreases during the transient as the steam pressure decreases. The steam release causes a decrease in the RCS temperature and the steam generator (SG) pressure. The decrease in the SG pressure results in the actuation of the low SG pressure trip signal, which trips the reactor. In the presence of a negative moderator temperature coefficient, the RCS temperature decreases and this results in an addition of positive reactivity. With the most reactive control rod assumed stuck in its fully withdrawn position after reactor trip, there is a possibility that the core may become critical and return to power, leading to a potential fuel failure in the core. The reactor is ultimately shut down because of the boric acid delivered by the safety injection system.

The licensee discussed the results of the MSLB analyses in Reference 2 and the licensee's September 28, 1998, letter. The licensee considered MSLB events with various combinations of initial plant conditions and evaluated the effects of break sizes, break locations (such as inside and outside containment, upstream and downstream of the isolation valves and the check valves in the steam lines), and thermal-hydraulic parameters and neutronic parameters on the MSLBs. The licensee identified the events that would result in fuel failure and analyzed those events to identify the most limiting MSLB cases. As a result, the licensee provided for the staff's review quantitative results of analyses for two categories of the MSLB events: pre-scram MSLB events and post-scram SLB events. For all cases, the licensee assessed fuel responses against the acceptable safety limits (Ref. 1) of the departure from nucleate boiling ratio (DNBR) and the fuel rod centerline melting (FCM).

2.1.1.a <u>Pre-scram MSLB Events</u>

To identify the limiting pre-scram cases with respect to the potential for fuel degradation, the licensee analyzed the following cases:

(1) MSLBs at full power (FP) outside containment and upstream of the main steamline check valves,

(2) MSLBs at FP outside containment and downstream of the main steamline check valves,

(3) MSLBs at FP inside containment and upstream of the main steamline check valves, and

(4) MSLBs at FP inside containment and upstream of the main steamline check valves with concurrent loss of offsite power (LOOP).

For pre-scram cases (cases 1 through 4), a full range of break sizes, up to the double-ended guillotine break of a main steamline, were considered. The MSLB cases were initiated from rated power (Including measurement uncertainties). At the rated power condition, the pre-scram power level is maximized, the stored energy in the primary RCS system is at highest levels and the available thermal margin is minimized. These conditions maximize the positive core reactivity feedback, core heat flux and, thus, maximize the potential for challenge to the safety DNBR and FCM limits for the pre-scram core. Other assumptions used in the analyses were use of the Doppler coefficient and moderator reactivity coefficient required by the TS to maximize the potential for the core to reach lowest margins to the safety DNBR and FCM limits. The reactor was assumed to trip on the signals from the low SG water level trip, low reactor coolant flow trip, variable overpower trip, thermal margin/low pressure trip and high containment pressure trip.

2.1.1.b. <u>Post-scram MSLB Events</u>

To identify the limiting cases with respect to the potential for post-scram return-to-power, the licensee analyzed MSLB cases both inside containment and outside containment with the following initial plant conditions:

(5) an MSLB at FP with concurrent LOOP,

(6) an MSLB at FP,

(7) an MSLB at zero power (ZP) with concurrent LOOP, and

(8) an MSLB at ZP.

The post-scram FP MSLB cases were initiated from rated power. At rated power conditions, the stored energy in the primary RCS system was at its highest levels, the available thermal margin was minimized and the pre-scram power level was maximized. These conditions resulted in the greatest potential for cooldown and provided the greatest challenge to the safety DNBR and FCM limits for the post-scram core. Thus, the MSLBs initiated from full power conditions bounded other cases initiated from lower power operation modes. For the FP MSLB cases, the reactor trip was assumed to occur on the low SG pressure trip signal.

The ZP MSLB cases were initiated from Mode 2. At Mode 2 conditions, the initial pressure, temperature and steam flow through the broken line were at their highest values compared to subcritical plant conditions (Modes 3 to 6.) An MSLB initiated from highest initial temperature and blowdown flow through the steam line provided the greatest potential for cooldown, thus, an MSLB initiated from Mode 2 bounded the MSLBs initiated from Modes 3 through 6.

For post-scram MSLBs (cases 5 though 8), the analyses were performed by assuming the largest possible size of the break, a double-ended rupture of a steam line upstream of the main steam isolation valve. This break was identified previously by the licensee as the limiting break, resulting in a greatest cooldown rate. The largest effective steam flow area for a steam line, which is limited by the integral steam generator flow restrictor throat area of 3.51 ft^2, was assumed in the analysis.

To maximize the return-to-power after the reactor trip, and, thus, maximize the potential for the fuel failure during the MSLBs, the following assumptions were used:

The most reactive control element assembly was assumed stuck in its fully withdrawn position after reactor trip.

For single failure considerations, the loss of one diesel generator (DG) was identified as the most limiting single failure. The loss of one DG resulted in the disabling of one of the two high pressure safety injection (HPSI) pumps required to be in service during normal operation. The assumption of the most limiting single failure reduced the boron flow injection rate and increased the potential for the return-to-power.

The HPSI system was modeled to take water from the RWST at 35 °F with a minimum boron concentration of 1720 parts per million required by the TS. The assumption of the low temperature and low boron concentration of the injected water minimized the boron effect and increased the potential for the return-to-power.

End-of-Cycle values for the required control rod shutdown worth and the TS moderator temperature coefficient were assumed in the analyses.

For the FP cases, the auxiliary feedwater (AFW) flow was assumed to be zero at break initiation. After 3 minutes (based on TS), AFW was delivered at the maximum capacity of the AFW system with flow restrictors installed on the AFW lines. For the ZP cases, the AFW was increased to the maximum capacity immediately at break initiation. For all cases, all of the AFW was directed to the affected SG to maximize the cooldown rate. The assumption of the maximum AFW to the affected SG maximized the cooldown rate and increased the potential for the return-to-power.

Since the assumptions discussed above maximize the positive core reactivity feedback, core heat flux and, thus, minimize the calculated minimum DNBR, the staff finds that the assumptions are conservative and acceptable.

2.1.2 Analytical Conclusions

The results of the analyses (see Attachment 5 of the licensee's September 28, 1998, letter and Table 4.1 of Ref. 2) identified the limiting MSLB cases. From the DNBR consideration, the limiting MSLB was Case (4): the pre-scram 3.51 ft^2 break at FP inside containment with concurrent LOOP, resulting in a minimum DNBR of 0.88 (which is below the 95/95 XNB correlation limit.) With the assumption that all fuel failed when they experienced DNBRs lower than the safety DNBR limits, the results of the analysis for Case (4) showed that 3.7 percent of the fuel rods in the core was predicted to fail because of low DNBRs. From the FCM consideration, the limiting event was Case (6): the post-scram MSLB at FP outside containment with offsite power available, resulting in a highest linear heat generation rate (LHGR) of 24.3 kW/ft. With the assumption that all fuel failed when they experienced LHGRs greater than safety limit of 21 kW/ft, the results of the analysis for Case (6) showed that one full fuel assembly, 0.5 percent of fuel in the core, was predicted to fail due to violation of the FCM limit.

Standard Review Plan (SRP) 15.1.5(II)(C)(2) specifies that for an acceptable MSLB analysis, (1) fuel failure must be assumed for all rods that do not meet the safety limits for fuel integrity (such as the safety DNBR limits) and (2) any fuel damage calculated to occur must be of sufficiently limited extent. The staff reviewed the calculated results of the MSLB analysis and found that the method consistent with the SRP is used for fuel failure determination and the calculated fuel damage limited to 3.7 percent of the fuel rods in the core is within the range previously approved by the staff for the MSLB analysis. The staff concludes that the analytical results satisfy the SRP 15.1.5(II)(C)(2) guidance and are, therefore, acceptable.

2.2 Radiological Effects of MSLB

NNECO revised the design-basis analysis for an MSLB at Millstone Unit 2. Two cases were considered, one involving the failure of a main steamline inside containment and the other involving the failure of a main steamline outside containment. NNECO postulated that 0.46 percent of the fuel rods in the core would be expected to fail. Two cases were evaluated for the radiological consequences of an MSLB outside containment. The first case evaluated the consequences of the MSLB assuming fission product release from the fuel rods postulated to fail. The second case evaluated the consequences of the MSLB assuming the occurrence of a preaccident iodine spike. Other analysis assumptions were tabulated in the proposed updated FSAR pages submitted with the amendment request. NNECO evaluated radiation doses at the exclusion area boundary, at the outer boundary of the low population, and in the control room. NNECO concluded that the radiological consequences for an MSLB would not exceed the guidelines of 10 CFR Part 100, and that the dose to Millstone Unit 2 control room operators would not exceed the 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-19 criteria.

The NRC staff reviewed the assumptions and inputs used by NNECO in its MSLB radiological analysis and found them acceptable. The NRC staff performed confirmatory calculations using these data. The results obtained by the NRC staff were comparable to those reported by NNECO. The NRC staff concludes that NNECO's analyses are acceptable. The analysis assumptions and inputs used by the NRC staff are tabulated in the attached Table 1. The NRC staff results are tabulated in the attached Table 2.

2.3 Radiological Effects of a LOCA

NNECO revised the design basis analysis for a LOCA at Millstone Unit 2 and submitted a description of the analysis and results obtained. Because of NRC staff concerns regarding a discrepancy between the secondary containment bypass leakage rates used in the analyses for offsite and control room doses, NNECO reanalyzed the radiological consequences of the design-basis accident (DBA) LOCA. A description of the updated analysis and the results obtained were submitted in a letter dated January 20, 1999. The analysis conservatively assumed that 100 percent of the core inventory of noble gases and 25 percent of the core inventory of radioiodine were instantaneously released to the containment atmosphere and were available for release to the environment. NNECO considered the following radioactivity release pathways:

> A portion of the airborne radioactivity in the primary containment is assumed to leak into the enclosure building where it collected, filtered, and released to the environment via the Millstone Unit 1 plant stack. For the first 110 seconds, during which the pressure in the enclosure building is being drawn down, the release from the containment is modeled as an unflitered ground level release.
>
> A small fraction of the leakage from the primary containment is assumed to bypass the enclosure building and to be released as an unflitered ground level release.
>
> A portion of the radioactivity in the containment sump is assumed to leak from systems that recirculate the sump water outside the containment. This release is collected, filtered, and released to the environment via the Millstone Unit 1 plant stack. The earliest that this recirculation would occur is 25 minutes.
>
> A small fraction of the radioactivity in the recirculated sump water is assumed to leak to the RWST. The RWST is vented to the atmosphere, providing a path for a portion of the radioactivity to escape to the environment. It is assumed that this leakage (approximately 0.2 gpm) does not reach the RWST for about 25 hours.

NNECO concluded that the radiological consequences for a LOCA would not exceed the guidelines of 10 CFR Part 100, and that the dose to Millstone Unit 2 control room operators would not exceed the 10 CFR Part 50, Appendix A, GDC-19 criteria.

The leakage assumed in the analysis of each release pathway is the applicable maximum allowable leakage provided in TS LCOs or other administrative controls. For this amendment request, NNECO has assumed a secondary containment bypass of 0.72 percent of the maximum allowable containment leakage. In the prior analyses, the bypass was assumed to be 1.7 percent for the offsite dose calculation and a lower value for the control room dose calculation. The revised bypass value applies to both dose calculations. By letter dated January 18, 1999, NNECO proposed to revise TS 3.6.1.2, *Containment Systems-Containment Leakage*, to reflect this reduced leakage.

The current FSAR describes an analysis of the radiological consequences of a post-LOCA hydrogen purge. The hydrogen purge system was included in the plant design as a backup to the installed hydrogen recombiners. NNECO is proposing to downgrade the hydrogen purge system and has omitted the evaluation of the radiological consequences of a hydrogen purge in the proposed reanalysis. Such a purge would be necessary only if both of the safety-grade hydrogen recombiners were to fail. The operability of these recombiners is provided for in the TS. It is beyond the design basis to assume that both of these recombiners fail. The NRC staff agrees with NNECO's proposal to omit consideration of radiation doses due to a hydrogen purge.

The current FSAR describes an analysis of the radiological consequences of a LOCA that occurs during high wind conditions. Under high wind conditions, the effectiveness of the secondary containment to collect primary containment leakage for filtration and release is degraded. A larger fraction of the primary containment leakage is assumed to bypass the secondary containment. The current FSAR analyses show that the radiation doses due to a LOCA during low wind speed conditions are more limiting than those postulated during high wind speeds. The improved atmospheric dispersion associated with the increased wind speed compensates for the increased unfiltered leakage. NNECO has proposed deleting the FSAR discussion of the high wind speed case. The NRC staff notes that the proposed changes to the assumptions and inputs used in the low wind speed case would not have affected the high wind speed case. The NRC staff agrees that the low wind speed case will remain limiting and that the high wind speed case discussion may be omitted.

The NRC staff reviewed the assumptions and inputs used by NNECO in its LOCA radiological analysis and found them acceptable. The NRC staff performed confirmatory calculations using these data. The results obtained by the NRC staff were comparable to those reported by NNECO. The NRC staff concludes that NNECO's analysis are acceptable. The analysis assumptions and inputs used by the NRC staff are tabulated in the attached Table 1. The NRC staff's results are tabulated in the attached Table 2.

2.4 <u>Control Room Doses</u>

NNECO states that all of the accidents at Millstone Unit 2 were re-analyzed for their effect on the doses in the Millstone Unit 2 control room and that of all the accidents, the MSLB was the most limiting. NNECO also considered the impact of a LOCA at Millstone Unit 3 on the

Millstone Unit 2 control room. NNECO had previously considered the impact of design basis accidents at Millstone Unit 1. Since Millstone Unit 1 is currently shutdown and the licensee has informed the NRC of its intent not to restart the unit, the Millstone Unit 1 analysis results have been omitted. Based on its review of the LOCA and MSLB materials submitted by NNECO and its experience with analyses of the other design-basis accidents, the NRC staff concludes that the Millstone Unit 2 control room doses would be within the acceptance criteria of 10 CFR Part 50, Appendix A, GDC-19, and NUREG-0800, Section 6.4.

2.5 Atmospheric Dispersion

NNECO proposed revised values for the atmospheric dispersion (χ/Q) assumed in the MSLB and LOCA analyses. The FSAR states that the χ/Q values are calculated using the methodology of Regulatory Guide 1.145[1] and Murphy-Campe.[2] NNECO provided confirmation that the revised values were calculated using these accepted methodologies. Based on NNECO's use of accepted methodologies and a qualitative review of the proposed values, the NRC staff concludes that the proposed values are acceptable for use in the MSLB and LOCA design-basis analyses.

2.6 Credit for Iodine Removal by Containment Spray System

The staff reviewed the licensee's proposal to credit the containment spray system for iodine removal based on SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System", Rev. 2, December 1988. In determining the elemental iodine removal coefficient, λ_s, the staff applied a model provided in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," June 1993. The results of the staff's evaluation confirm the licensee's use of the maximum allowable λ_s of 20 per hour, as given in the SRP. In addition, the staff verified the particulate iodine removal coefficient, λ_p, of 3.03 per hour.

The staff also reviewed the licensee's calculated Decontamination Factor (DF), which supports the use of the maximum value of 200, as allowed in the SRP. In determining DF, the staff applied data provided in NUREG/CR-4697, "Chemistry and Transport of Iodine in Containment," October 1986, to determine the effective iodine partition coefficient. The results of the staff's evaluation verify the licensee's use of 200 for the DF.

Based on its evaluation, the staff finds that the following chemical-related parameters for Millstone Unit 2 containment spray system are consistent with acceptable effective fission product removal and retention during post-accident conditions:

$$\lambda_s = 20 \text{ per hour}, \lambda_p = 3.03 \text{ per hour, and DF} = 200.$$

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1 USNRC, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. Regulatory Guide 1.145

2 Murphy, K.G. and Campe, K.W., *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19*, published in proceedings of 13th AEC Air Cleaning Conference

Therefore, the staff concludes that the licensee's proposal to credit the containment spray system for iodine removal is acceptable.

2.6 TS Changes

2.6.1 TS 3.3.2.1, *Instrumentation-Engineered Safety Features Actuation System*

NNECO proposed to correct spelling errors and add a historical amendment number to one TS page. These are acceptable editorial corrections.

2.6.2 TS 3.4.6.2, *Reactor Coolant System-Reactor Coolant System Leakage*

NNECO proposed to reduce the maximum allowable primary-to-secondary leakage from the current 1.0 gpm to 0.035 gpm (per steam generator). The proposed TS is consistent with the leakage assumptions made in the revised MSLB analysis. Analyses for other DBAs that are based in part on allowable primary-to-secondary leakage of 1.0 gpm continue to be bounding. Since NNECO is proposing to reduce the maximum leakage, the NRC staff concludes that there is reasonable assurance that there will be no increase in the consequences or probability of any previously analyzed accident because of these changes. The NRC staff finds that the proposed changes to Surveillance Requirements 4.4.6.2.1 and 4.4.6.2.2 do not change the intent of the specification and will have no adverse impact on plant operations.

2.6.3 TS 3.4.8, *Reactor Coolant System-Specific Activity*

NNECO's proposed changes clarify provisions regarding specific activity LCOs and surveillance testing. The NRC staff finds that the proposed revisions do not change the intent of the TS. The NRC staff concludes that there is reasonable assurance that there will be no increase in the consequences or probability of any previously analyzed accident because of these changes.

2.6.4 TS 3.6.2.1, *Containment Systems-Depressurization and Cooling Systems Containment Spray and Cooling Systems*

The licensee's revised radiological assessment calculation for the design-basis LOCA credits iodine removal from the containment atmosphere by the core containment spray system. This reduced allowed outage time is consistent with NUREG-1432, and is therefore acceptable.

2.6.5 TS 3.6.5.1, *Containment Systems-Secondary Containment Enclosure Building Filtration System*

The licensee's allowed pressure drop across the high-efficiency particulate air filters and charcoal absorber banks specified in TS 4.6.5.1.d.1 will be reduced from $\leq$ 6 inches water gauge to $\leq$ 2.6 inches water gauge. The new value is plant specific. Since the licensee is replacing a generic value with a plant-specific value that is more restrictive, the licensee's proposal is acceptable.

2.6.5 TS 3.7.6.1, *Plant Systems-Control Room Emergency Ventilation System*

TS 3.7.6.1 was revised to state that the two control room emergency ventilation trains shall be operable in all modes. The licensee proposed to replace "system" with "train," "air clean-up system" with "ventilation train," and "control air conditioning system" with "control room emergency ventilation system" for the entire TS 3.7.6.1. The licensee justified the changes as to standardize the terminology used throughout the specification. The staff finds the licensee's proposal acceptable because it clarifies the TS.

TS 4.7.6.1.e.1 was revised to state that at least once per 18 months, verify that the pressure drop across the combined HEPA filters and charcoal adsorber bank is less than 3.4 inches water gauge (changed from current 6 inches water gauge) while operating the train at a flow rate of 2500 cfm ±10%. The licensee's basis for the change is that the current value is a generic value and the proposed value is a plant-specific and more restrictive value. Since the licensee is replacing a generic value with a plant-specific value that is more restrictive, the licensee's proposal is acceptable.

TS 4.7.6.1.e.2 was revised to state that at least once per 18 months, verify that on a recirculation signal, with the control room emergency ventilation (CREV) train operating in normal mode and the smoke purge mode, the train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks. The licensee proposed to add the phrase "with the control room emergency ventilation train operating in the normal mode and the smoke purge mode," to the surveillance requirement to require that the recirculation mode be the only operating mode that the CREV system can be in before receiving a recirculation actuation signal. The licensee's failure to test whether the recirculation actuation signal overrides the smoke purge actuation signal was identified in the NRC Inspection Report 50-336/95-201 as Deficiency 95-201-02. The surveillance procedure was modified to address this issue. The licensee's basis for this change is to establish the initial conditions necessary for verification of the CREV system operation. Based on the staff's review, the proposed changes will clarify the requirements of the CREV system and are justified by the information provided by the licensee.

TS 4.7.6.1.e.3 and FSAR Section 9.9.10.3.2 were revised to increase the maximum allowable control room air in-leakage from 100 scfm to 130 scfm. NNECO stated that all of the DBAs at Millstone Unit 2 were re-analyzed to assess the effect of this increased in-leakage on the doses postulated for the Millstone Unit 2 control room. NNECO also considered the impact of a LOCA at Millstone Unit 3 on the Millstone Unit 2 control room assuming the increased in-leakage. As discussed in Section 2.4 of the SE, the NRC staff concludes that there is reasonable assurance that the Millstone Unit 2 control room doses would continue to meet the acceptance criteria of 10 CFR Part 50, Appendix A, GDC-19, and NUREG-0800, Section 6.4, with the increased in-leakage.

2.6.6 TS 3.9.15, *Refueling Operations-Storage Pool Area Ventilation System-Fuel Storage*

The licensee's allowed pressure drop across the high-efficiency particulate air filters and charcoal absorber banks specified in TS 4.9.15.d.1 will be reduced from ≤ 6 inches water gauge to ≤ 2.6 inches water gauge. The new value is plant-specific. Since the licensee is replacing a generic value with a plant-specific value that is more restrictive, the licensee's proposal is acceptable.

2.6.7 TS 6.9.1.8, *Core Operating Limits Report*

TS 6.9.1.8b lists the references documenting the analytical methods that are used by the licensee to perform safety analyses. The licensee's TS changes involve changes updating the references specified in TS 6.9.1.8b. The following are the proposed TS changes and the staff's evaluation:

(1) Add clarifications and specific revision numbers to current references 1, 2, 3, 5, 7, 8 and 9 in TS 6.9.1.8b. Reference 10 is part of the current references since reference 6 has been split into reference 6 and reference 7 in the revised list. The changes are acceptable since the analytical methodologies remain unchanged for these references.

(2) Add references 11, 12, 13, 14, and 15 to proposed TS 6.9.1.8b for completeness. The added references are NRC-approved topical reports documenting the SPC methodologies used by the licensee to determine the core operating limits. Therefore, the TS changes are acceptable.

(3) Change reference 4 in TS 6.9.1.8b to reflect a revised MSLB analysis methodology. The revised methodology (Ref. 1) is acceptable for the Millstone Unit 2 analysis. The TS change is acceptable.

(4) Remove the sentence on page 6-19 of the TS that starts with "The Acceptable Millstone 2..:" and ends with "...dated October, 1988." The TS change involves a removal of an outdated reference documenting the plant-specific MSLB analysis, and is acceptable.

Therefore, the NRC staff concludes that there is reasonable assurance that there will be no increase in the consequences or probability of any previously analyzed accident because of these changes.

2.7 TS Bases and FSAR Changes

The licensee proposed TS Bases and FSAR changes corresponding with the preceding TS changes. The staff found that the proposed changes provided appropriate information in support of this license amendment request. Therefore, the licensee's proposed TS Bases and FSAR changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 53951, October 7, 1998, and 63 FR 66597, December 2, 1998). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has 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, (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.

Attachments: 1. Table 1
2. Table 2

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Date: March 10, 1999

REFERENCES

1. EMF-84-093(P), Revision 1, Main Steamline Break Methodology for Millstone 2, dated June 1998.

2. EMF-98-036, Revision 1, Post-Scram Main Steamline Break Analysis for Millstone Unit 2, dated July 1998.

Table 1

Accident Analysis Parameters Used by NRC staff

Common to LOCA and MSLB

Reactor power, MWt	2754
Dose conversion factors	FGR-11 & FGR-12
Offsite breathing rates, m^3/sec	
0-8 hrs	3.47E-4
8-24 hrs	1.75E-4
24-720 hrs	2.32E-4

Meteorology, sec/m^3

				---- Control Room ----		
	Time	Elevated	Ground	Elevated	Ground	MSLB
EAB	0-2 hrs	1.00E-4	3.66E-4			
LPZ	0-4 hrs	2.69E-5	4.80E-5	2.51E-4	3.07E-3	3.19E-3
	4-8 hrs	3.04E-6	2.31E-5	1.96E-4	3.07E-3	3.19E-3
	8-24 hrs	2.17E-6	1.60E-5	5.46E-6	2.09E-3	2.85E-3
	24-96 hrs	1.04E-6	7.25E-6	2.06E-7	7.42E-4	1.12E-3
	96-720 hrs	3.63E-7	2.32E-6	2.58E-9	1.93E-4	3.63E-4

Control room volume, ft^3	35,650
Control room intake prior to isolation, cfm	800
Control room inleakage during isolation, cfm	130
Control room recirculation flow, cfm	2250
Control room recirculation filter efficiency, %	90
Control room isolation, sec	
LOCA	5
MSLB	90
Control room recirculation filter delay, minutes	10
Control room breathing rate, m^3/sec	3.47E-4

Loss of Coolant Accident

Core inventory release to primary containment, %	
Iodine	25
Noble Gases	100

Iodine species	
Elemental	91
Organic	4
Particulate	5
Primary containment volume, ft^3	1.9E6
Primary containment leakage, %/day	
0-24 hours	0.5
24-720 hours	0.25
Secondary containment bypass as % of primary containment leakage	0.72
Primary containment spray efficiency, hr^{-1}	
Elemental iodine	20
Particulate iodine	3.03
Fraction of containment sprayed	0.75
Spray actuation, sec	101
Spray duration, hours	
Elemental	0.715
Particulate	1.58
Spray decontamination factor	
Elemental	100
Particulate	25
Containment mixing, number of turnovers of unsprayed region	
101-303 seconds	6.06
303-454 seconds	7.76
454 seconds to 1.58 hours	6.34
Containment sump volume, gal	286,000
Fraction of core inventory iodines in sump	0.5
Leakage from ECCS systems outside containment, gph x 2	24
Leakage flash fraction	0.1
Start of ECCS leakage, minutes	25
RWST backleakage leakage, gpm	
0-25.45 hrs	0.0
25.45-27.70 hrs	0.01
27.70-28.37 hrs	0.13
28.37-29.60 hrs	0.14
after 29.60 hrs	0.19
Iodine DF (mass basis) in RWST	100

Enclosure building filtration efficiency, %	
Elemental	90
Particulate	90
Organic	70
Enclosure building drawdown time, sec	110

LOCA Release point

Containment airborne release prior to 110 sec	Ground
Containment airborne release after 110 sec	Elevated
Containment bypass release	Ground
ECCS leakage	Elevated
RWST leakage	Ground

Main Steam Line Break

Fuel melt fraction	0.0046
Fuel peaking factor	1.45
RCS liquid mass, lb	430,000
Intact SG liquid minimum mass, lb/SG	100,000
Primary-to-secondary leakage, gpm	0.035
Density of primary-to-secondary leakage, gm/cm^3	1.0
RCS activity*, µCi/gm DE I-131	
Case 1	1.0
Case 2	60.0

**FSAR Table 11.A-1 activities normalized to specified DE I-131 values using ICRP-30 DCFs*

Initial secondary activity, µCi/gm DE I-131	0.1

Environmental release basis

- Faulted SG blows down in 750 seconds, releasing all initial secondary activity
- Intact SG blows down through MSLB until isolated, 20 seconds
- Primary-to-secondary leakage into faulted SG for 24 hours, no holdup in SG
- Primary-to-secondary leakage into intact SG for 16 hours, iodine reduced by 0.01

Table 2

NRC Staff Confirmatory Analysis Results, rem

Main Steam Line Break

	Result	Acceptance* Criteria
Exclusion Area Boundary, 0-2 hours		
Thyroid	3.8	300.0
Whole Body	0.1	25.0
Low Population Zone, 30 days		
Thyroid	2.0	300.0
Whole Body	0.04	25.0
Control Room, 30 days		
Thyroid	22.0	30.0
Whole Body	0.009	5.0

Loss of Coolant Accident

	Result	Acceptance Criteria
Exclusion Area Boundary, 0-2 hours		
Thyroid	38.0	300.0
Whole Body	4.3	25.0
Low Population Zone, 30 days		
Thyroid	13.0	300.0
Whole Body	1.8	25.0
Control Room, 30 days		
Thyroid	25.0	30.0
Whole Body	0.3	5.0

**Since fuel damage was projected, the full 10 CFR Part 100 dose guidelines apply.*