Mr. R. P. Necci - Vice President
Nuclear Oversight and Regulatory Affairs
Northeast Nuclear Energy Company
c/o Mr. David A. Smith
P.O. Box 128
Waterford, CT 06385

June 29, 1999

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 RE: ISSUANCE OF AMENDMENT (TAC NO. MA4460)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 236 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated .

The amendment changes Technical Specifications 3.5.2, "Emergency Core Cooling Systems - ECCS Subsystems - Tavg≥ 300 °F;" 3.6.2.1, "Containment Systems - Depressurization and Cooling Systems - Containment Spray and Cooling Systems;" 3.7.1.2, "Plant Systems - Auxiliary Feedwater Pumps;" 3.7.3.1, "Plant Systems - Reactor Building Closed Cooling Water System;" and 3.7.4.1, "Plant Systems - Service Water System." The Bases of the associated Technical Specifications will be modified to address the proposed changes.

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,

Original signed by:

Ronald B. Eaton, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 236 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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Docket File	REaton	GHill(2)	JColaccino
PUBLIC	TClark	R. Scholl (SE to RFS)	
PDI-2 Reading	OGC	JDurr, RGN-I	
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EAdensam (EGA1)	WBeckner	DTerao	

DF01

C10031

*See previous concurrence

OFFICE	PDI-2/PM	PDI-2/LA	EMEB/BC*	OGC*	PDI-2/SC
NAME	REaton/vw	TClark	KManoly	RBachman	JClifford
DATE	6/25/99	6/25/99	03/17/99	03/25/99	6/25/99

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CP-1

Mr. R. P. Necci - Vice President
Nuclear Oversight and Regulatory Affairs
Northeast Nuclear Energy Company
c/o Mr. David A. Smith
P.O. Box 128
Waterford, CT 06385

June 29, 1999

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 RE: ISSUANCE OF AMENDMENT (TAC NO. MA4460)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 236 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated .

The amendment changes Technical Specifications 3.5.2, "Emergency Core Cooling Systems - ECCS Subsystems - Tavg≥ 300 °F;" 3.6.2.1, "Containment Systems - Depressurization and Cooling Systems - Containment Spray and Cooling Systems;" 3.7.1.2, "Plant Systems - Auxiliary Feedwater Pumps;" 3.7.3.1, "Plant Systems - Reactor Building Closed Cooling Water System;" and 3.7.4.1, "Plant Systems - Service Water System." The Bases of the associated Technical Specifications will be modified to address the proposed changes.

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Ronald B. Eaton, Sr. Project Manager, Section 2
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*See previous concurrence

OFFICE	PDI-2/PM	PDI-2/LA	EMEB/BC*	OGC*	PDI-2/SC
NAME	REaton/vw	TClark	KManoly	RBachman	JClifford
DATE	6/25/99	6/25/99	03/17/99	03/25/99	6/25/99

OFFICIAL RECORD COPY
DOCUMENT NAME: MI4660.AMD

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

June 29, 1999

Mr. R. P. Necci - Vice President
Nuclear Oversight and Regulatory Affairs
Northeast Nuclear Energy Company
c/o Mr. David A. Smith
P.O. Box 128
Waterford, CT 06385

SUBJECT: ISSUANCE OF AMENDMENT - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2 (TAC NO. MA4660)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No. 236 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated January 4, 1999, as supplemented April 7, 1999.

The amendment changes Technical Specifications 3.5.2, "Emergency Core Cooling Systems - ECCS Subsystems - Tavg≥ 300 °F;" 3.6.2.1, "Containment Systems - Depressurization and Cooling Systems - Containment Spray and Cooling Systems;" 3.7.1.2, "Plant Systems - Auxiliary Feedwater Pumps;" 3.7.3.1, "Plant Systems - Reactor Building Closed Cooling Water System;" and 3.7.4.1, "Plant Systems - Service Water System." The Bases of the associated Technical Specifications will be modified to address the proposed changes.

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,

Ronald B. Eaton, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 236 to DPR-65
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

cc:

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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. 236
License No. DPR-65

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 January 4, 1999, as supplemented April 7, 1999, 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. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 236 , 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, and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

M Bagwis for

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 29, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 236

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Remove	Insert
3/4 5-4	3/4 5-4
3/4 6-12	3/4 6-12
3/4 7-4	3/4 7-4
3/4 7-11	3/4 7-11
3/4 7-12	3/4 7-12
B 3/4 5-2	B 3/4 5-2
B 3/4 5-2a	B 3/4 5-2a
--	B 3/4 5-2b
B 3/4 6-3	B 3/4 6-3
B 3/4 6-3a	B 3/4 6-3a
B 3/4 6-3b	B 3/4 6-3b
B 3/4 6-3c	B 3/4 6-3c
B 3/4 7-2	B 3/4 7-2
B 3/4 7-3a	B 3/4 7-3a
B 3/4 7-4	B 3/4 7-4
--	B 3/4 7-4a

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

1. Verifying that each high-pressure safety injection pump:

a) Starts automatically on a test signal.

b) Develops a differential pressure of ≥ 1193 psid on recirculation flow.

c) Operates for at least 15 minutes.

2. Verifying that each low-pressure safety injection pump:

a) Starts automatically on a test signal.

b) Develops a differential pressure of ≥ 163 psid on recirculation flow.

c) Operates for at least 15 minutes.

3. Verifying that each charging pump:

a) Starts automatically on a test signal.

b) Operates for at least 15 minutes.

4. Verifying that each boric acid pump (when required OPERABLE per Specification 3.5.2.d):

a) Starts automatically on a test signal.

b) Develops a discharge pressure of ≥ 98 psig on recirculation flow.

c) Operates for at least 15 minutes.

5. Verifying that upon a sump recirculation actuation signal, the containment sump isolation valves open.

6. Cycling each testable, automatically operated valve through at least one complete cycle.

7. Verifying the correct position for each manual valve not locked, sealed or otherwise secured in position.

8. Verifying the correct position for each remote or automatically operated valve.

9. Verifying that each ECCS subsystem is aligned to receive electrical power from separate OPERABLE emergency busses.

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 differential pressure of ≥ 232 psid,

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

AUXILIARY FEEDWATER PUMPS

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps shall be OPERABLE with:

a. Two feedwater pumps capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. Entry into an OPERATIONAL MODE or other specified condition under the provisions of Specification 3.0.4 shall not be made with three auxiliary feedwater pumps inoperable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each pump from the control room,

2. Verifying that:

a) Each motor driven pump develops a differential pressure of $\geq$ 1144 psid on recirculation flow, and

b) The steam turbine driven pump develops a differential pressure of $\geq$ 1113 psid, corrected to rated pump speed, on recirculation flow when the secondary steam supply pressure is greater than 800 psig. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 Two independent reactor building closed cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one reactor building closed cooling water loop inoperable, restore the inoperable loop to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 Each reactor building closed cooling water loop shall be demonstrated OPERABLE:

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

   1. Starting (unless already operating) each pump from the control room,

   2. Verifying that each pump develops at least 93% of the differential pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.

   3. Verifying that each pump operates for at least 15 minutes,

   4. Verifying that each loop is aligned to receive electrical power from separate OPERABLE emergency busses.

   5. Verifying correct position of all valves servicing safety related equipment that are not locked, sealed or otherwise secured in position, and

   6. Exercising all automatically operated valves servicing safety related equipment and testable during plant operation.

b. At least once per 18 months by exercising all power operated valves through one complete cycle of full travel.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

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

1. Starting (unless already operating) each pump from the control room,

2. Verifying that each pump develops at least 93% of the differential pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.

3. Verifying that each pump operates for at least 15 minutes,

4. Verifying that each loop is aligned to receive electrical power from separate OPERABLE emergency busses.

5. Verifying correct position of all valves servicing safety related equipment that are not locked, sealed or otherwise secured in position, and

6. Exercising all automatically operated valves servicing safety related equipment and testable during plant operation.

b. At least once per 18 months* by exercising all power operated valves through one complete cycle of full travel.

---

*Except that the surveillance requirement due no later than May 5, 1994, may be deferred until the next refueling outage, but no later than September 30, 1994, whichever is earlier.

3/4.5.1 SAFETY INJECTION TANKS (continued)

within 6 hours and pressurizer pressure reduced to < 1750 psia within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner and without challenging plant systems.

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward.

The ECCS leak rate surveillance requirements assure that the leakage rates assumed for the system outside containment during the recirculation phase will not be exceeded.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. The purpose of the HPSI and LPSI pumps differential pressure test on recirculation ensures that the pump(s) have not degraded to a point where the accident analysis would be adversely impacted.

The acceptance criteria for the HPSI pumps Technical Specification Surveillance Requirement (SR 4.5.2.a.1.b), a minimum pump recirculation flow test, was developed assuming a 5% degraded pump using the manufacturer curves. The associated accident analyses assume a HPSI flow that represents 5% degradation. Early delivery of HPSI pump flow, at high head conditions similar to those established when the pump is on recirculation flow, is an important assumption in the accident analyses. Flow measurement instrument inaccuracy has been accounted for in the design basis hydraulic analysis. Pressure measurement instrument inaccuracy will be accounted for in the acceptance criteria contained in the surveillance procedure for SR 4.5.2.a.1.b. Pressure measurement instrument inaccuracy is not reflected in the Technical Specification acceptance criteria.

The acceptance criteria for the LPSI pumps Technical Specification Surveillance Requirement (SR 4.5.2.a.2.b) was developed assuming a 10% degraded pump from the actual pump curves. The associated accident analyses assume a LPSI flow that represents 10% degradation. For the limiting large

break loss of coolant accident (LBLOCA) analysis case, the analysis does not credit LPSI flow following the safety injection actuation signal until after a time delay which simulates the time for the emergency diesel generators to start and load. After this delay, the Reactor Coolant System (RCS) has depressurized well below the shutoff head of the LPSI pumps. At this low RCS pressure, the operating point of the pumps is significantly greater than minimum recirculation flow. For boron precipitation control following a loss of coolant accident, the LPSI pump is credited with providing hot leg injection flow. The operating point for the LPSI pumps during hot leg injection is also greater than minimum recirculation flow. Flow measurement instrument inaccuracy has been accounted for in the design basis hydraulic analysis. Pressure measurement instrument inaccuracy will be applied and controlled by the surveillance procedures when verifying pump performance in the flow ranges credited in the accident analyses. No correction for pressure measurement instrument inaccuracy will be applied to minimum recirculation flow type test data since this portion of the curve is not credited in the accident analyses. Pressure measurement instrumentation inaccuracy is not reflected in either Technical Specification SR 4.5.2.a.2.b, or in the associated surveillance procedure.

The purpose of the ECCS throttle valve surveillance requirements is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Verification of the correct position for the mechanical and/or electrical valve stops can be performed by either of the following methods:

1. Visually verify the valve opens to the designated throttled position; or

2. Manually position the valve to the designated throttled position and verify that the valve does not move when the applicable valve control switch is placed to "OPEN."

In MODE 4 the automatic safety injection signal generated by low pressurizer pressure and high containment pressure and the automatic sump recirculation actuation signal generation by low refueling water storage tank level are not required to be OPERABLE. Automatic actuation in MODE 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Since the manual actuation (trip pushbuttons) portion of the safety injection and sump recirculation actuation signal generation is required to be OPERABLE in MODE 4, the plant operators can use the manual trip pushbuttons to rapidly position all components to the required accident position. Therefore, the safety injection and sump recirculation actuation trip pushbuttons satisfy the requirement for generation of safety injection and sump recirculation actuation signals in MODE 4.

In MODE 4, the OPERABLE HPSI pump is not required to start automatically on a SIAS. Therefore, the pump control switch for this OPERABLE pump may be placed in the pull-to-lock position without affecting the OPERABILITY of the pump. This will prevent the pump from starting automatically, which could result in overpressurization of the Shutdown Cooling System. Only one HPSI pump may be OPERABLE in MODE 4 with RCS temperatures less than or equal to 275°F due to the restricted relief capacity with Low-Temperature Overpressure Protection System. To reduce shutdown risk by having additional pumping capacity readily available, a HPSI pump may be made inoperable but available at short notice by shutting its discharge valve with the key lock on the control panel.

The provision in Specification 3.5.3 that Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 4 is provided to allow for connecting the HPSI pump breaker to the respective power supply or to remove the tag and open the discharge valve, and perform the subsequent testing necessary to declare the inoperable HPSI pump OPERABLE. Specification 3.4.9.3 requires all HPSI pumps to be not capable of injecting into the RCS when RCS temperature is at or below 190°F. Once RCS temperature is above 190°F one HPSI pump can be capable of injecting into the RCS. However, sufficient time may not be available to ensure one HPSI pump is OPERABLE prior to entering MODE 4 as required by Specification 3.5.3. Since Specifications 3.0.4 and 4.0.4 prohibit a MODE change in this situation, this exemption will allow Millstone Unit No. 2 to enter MODE 4, take the steps necessary to make the HPSI pump capable of injecting into the RCS, and then declare the pump OPERABLE. If it is necessary to use this exemption during plant heatup, the appropriate action statement of Specification 3.5.3 should be entered as soon as MODE 4 is reached.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) after a LOCA the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the control element assembly (CEA) of highest worth, which remains withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core.

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 $\geq$ 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.

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and the subsystem OPERABILITY is maintained. The purpose of the containment spray pumps differential pressure test on recirculation, Surveillance Requirement 4.6.1.1.a.2, ensures that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the containment spray pumps was developed assuming a 5% degraded pump from the actual pump curves. Flow and pressure measurement instrument inaccuracies have been accounted for in the design basis hydraulic analysis. It is not necessary to account for either flow or pressure measure instrument inaccuracy in the acceptance criteria contained in the surveillance procedure. Flow and

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS (Continued)

pressure measurement instrument inaccuracies are already reflected in the Technical Specification acceptance criteria.

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 the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment isolation valves are used to close all fluid (liquid and gas) penetrations not required for operation of the engineered safety feature systems, to prevent the leakage of radioactive materials to the environment. The fluid penetrations which may require isolation after an accident are categorized as Type P, O, or N. The penetration types are listed with the containment isolation valves in the Technical Requirements Manual.

Type P penetrations are lines that connect to the reactor coolant pressure boundary (Criterion 55 of 10CFR50, Appendix A). These lines are provided with two containment isolation valves, one inside containment, and one outside containment.

Type O penetrations are lines that are open to the containment internal atmosphere (Criterion 56 of 10CFR50, Appendix A). These lines are provided with two containment isolation valves, one inside containment, and one outside containment.

Type N penetrations are lines that neither connect to the reactor coolant pressure boundary nor are open to the containment internal atmosphere, but do form a closed system within the containment structure (Criterion 57 of 10CFR50, Appendix A). These lines are provided with single containment isolation valves outside containment. These valves are either remotely operated or locked closed manual valves.

Locked or sealed closed containment isolation valves may be opened on an intermittent basis provided appropriate administrative controls are established. The position of the NRC concerning acceptable administrative controls is contained in Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," and includes the following considerations:

(1) stationing an operator, who is in constant communication with the control room, at the valve controls,

(2) instructing this operator to close these valves in an accident situation, and

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

(3) assuring that environmental conditions will not preclude access to close the valve and that this action will prevent the release of radioactivity outside the containment.

The appropriate administrative controls, based on the above considerations, to allow locked or sealed closed containment isolation valves to be opened are contained in the procedures that will be used to operate the valves. Entries should be placed in the Shift Manager Log when these valves are opened and closed. However, it is not necessary to log into any Technical Specification Action Statement for these valves, provided the appropriate administrative controls have been established.

If a locked or sealed closed containment isolation valve is opened while operating in accordance with Abnormal or Emergency Operating Procedures (AOPs and EOPs), it is not necessary to establish a dedicated operator. The AOPs and EOPs provide sufficient procedural control over the operation of the containment isolation valves.

Opening a locked or sealed closed containment isolation valve bypasses a plant design feature that prevents the release of radioactivity outside the containment. Therefore, this should not be done frequently, and the time the valve is opened should be minimized. As a general guideline, a locked or sealed closed containment isolation valve should not be opened longer than the time allowed to restore the valve to OPERABLE status, as stated in the action statement for LCO 3.6.3.1 "Containment Isolation Valves."

A discussion of the appropriate administrative controls for the containment isolation valves, that are expected to be opened during operation in MODES 1 through 4, is presented below.

Manual containment isolation valve 2-SI-463, safety injection tank (SIT) recirculation header stop valve, is opened to fill or drain the SITs and for Shutdown Cooling System (SDC) boron equalization. While 2-SI-463 is open, a dedicated operator, in continuous communication with the control room, is required.

When SDC is initiated, SDC suction isolation remotely operated valves 2-SI-652 and 2-SI-651 (inside containment isolation valve) and manual valve 2-SI-709 (outside containment isolation valve) are opened. 2-SI-651 is normally operated from the control room. While in Modes 1, 2 or 3, 2-SI-651 is closed with the closing and opening coils removed and stored to satisfy Appendix R requirements. It does not receive an automatic containment isolation closure signal, but is interlocked to prevent opening if Reactor Coolant System (RCS) pressure is greater than approximately 275 psia. When 2-SI-651 is opened from the control room, either one of the two required licensed (Reactor Operator) control room operators can be credited as the dedicated operator required for administrative control. It is not necessary to use a separate dedicated operator.

When valve 2-SI-709 is opened locally, a separate dedicated operator is not required to remain at the valve. 2-SI-709 is opened before 2-SI-651. Therefore, opening 2-SI-709 will not establish a connection between the RCS and the SDC System. Opening 2-SI-651 will connect the RCS and SDC System. If a problem then develops, 2-SI-651 can be closed from the control room.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

The administrative controls for valves 2-SI-651 and 2-SI-709 only apply during SDC operation. They are acceptable because RCS pressure and temperature are significantly below normal operating pressure and temperature (the RCS is administratively required to be < 300 °F and < 265 psia before shutdown cooling flow is initiated), the penetration flowpath can be isolated from the control room by closing either 2-SI-652 or 2-SI-651, and the manipulation of these valves, during this evolution, is controlled by plant procedures.

The pressurizer auxiliary spray valve, 2-CH-517, can be used as an alternate method to decrease pressurizer pressure, or for boron precipitation control following a loss of coolant accident. When this valve is opened from the control room, either one of the two required licensed (Reactor Operator) control room operators can be credited as the dedicated operator required for administrative control. It is not necessary to use a separate dedicated operator.

The exception for 2-CH-517 is acceptable because the fluid that passes through this valve will be collected in the Pressurizer (reverse flow from the Pressurizer to the charging system is prevented by check valve 2-CH-431), and the penetration associated with 2-CH-517 is open during accident conditions to allow flow from the charging pumps. Also, this valve is normally operated from the control room, under the supervision of the licensed control room operators, in accordance with plant procedures.

A dedicated operator is not required when opening remotely operated valves associated with Type N fluid penetrations (Criterion 57 of 10CFR50, Appendix A). Operating these valves from the control room is sufficient. The main steam isolation valves (2-MS-64A and 64B), atmospheric steam dump valves (2-MS-190A and 190B), and the containment air recirculation cooler RBCCW discharge valves (2-RB-28.2A-D) are examples of remotely operated containment isolation valves associated with Type N fluid penetrations.

MSIV bypass valves 2-MS-65A and 65B are remotely operated MOVs, but while in MODE 1, they are closed with their opening coils removed and stored to satisfy Appendix "R" requirements.

Local operation of the atmospheric steam dump valves (2-MS-190A and 190B), or other remotely operated valves associated with Type N fluid penetrations, will require a dedicated operator in constant communication with the control room, except when operating in accordance with AOPs or EOPs. Even though these valves can not be classified as locked or sealed closed, the use of a dedicated operator will satisfy administrative control requirements. Local operation of these valves with a dedicated operator is equivalent to the operation of other manual (locked or sealed closed) containment isolation valves with a dedicated operator.

The main steam supplies to the turbine driven auxiliary feedwater pump (2-MS-201 and 2-MS-202) are remotely operated valves associated with Type N fluid penetrations. These valves are maintained open during power operation. 2-MS-201 is maintained energized, so it can be closed from the control room, if necessary, for containment isolation. However, 2-MS-202 is deenergized

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of off-site power.

Any single motor driven or steam driven pump has the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 300°F where the shutdown cooling system may be placed into operation for continued cooldown.

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the auxiliary feedwater pumps differential pressure tests on recirculation, Surveillance Requirements 4.7.1.2.a.2.a and 4.7.1.2.a.2.b, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the motor driven auxiliary feedwater pumps was developed assuming a 5% degraded pump from the actual pump curves. The surveillance requirement acceptance criteria for the turbine driven auxiliary feedwater pump was developed from high flow test data extrapolated to minimum recirculation flow, and can be adjusted to account for the affect on pump performance of variations in pump speed. Flow and pressure measurement instrument inaccuracies have not been accounted for in the design basis hydraulic analysis for the motor driven auxiliary feedwater pumps. Flow, pressure, and speed measurement instrument inaccuracies have not been accounted for in the design basis hydraulic analysis for the turbine driven auxiliary feedwater pump. Corrections for flow, pressure, and speed (turbine driven pump only) measurement instrument inaccuracies will be applied to test data taken when verifying pump performance in the flow ranges credited in the accident analyses. No corrections for flow, pressure, and speed (turbine driven pump only) measurement instrument inaccuracies will be applied to minimum recirculation flow type test data since this portion of the curve is not credited in the accident analyses. Corrections for flow, pressure, and speed (turbine driven pump only) measurement instrument inaccuracies are not reflected in the Technical Specification acceptance criteria.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 300°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to atmosphere. The contained water volume limit includes an allowance for water not usable due to discharge nozzle pipe elevation above tank bottom, plus an allowance for vortex formation.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction

a feedwater isolation signal since the steam line break accident analysis credits them in prevention of feed line volume flashing in some cases. Feedwater pumps are assumed to trip immediately with an MSI signal.

3/4.7.1.7 ATMOSPHERIC STEAM DUMP VALVES

The atmospheric steam dump valves (ASDVs) provide a method for maintaining the unit in HOT STANDBY, and to cool the unit to Shutdown Cooling (SDC) System entry conditions if heat removal by the condenser steam dump valves is not available. The ASDVs are normally operated from the main control room. Local manual operation of the ASDVs is provided. The ASDVs are OPERABLE as long as the valves can be opened from the control room, or locally at the valves.

3/4.7.1.8 STEAM GENERATOR BLOWDOWN ISOLATION VALVES

The steam generator blowdown isolation valves will isolate steam generator blowdown on low steam generator water level. An auxiliary feedwater actuation signal will also be generated at this steam generator water level. Isolation of steam generator blowdown will conserve steam generator water inventory following a loss of main feedwater. The steam generator blowdown isolation valves will also close automatically upon receipt of a containment isolation signal or a high radiation signal (steam generator blowdown or condenser air ejector discharge).

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

The OPERABILITY of the reactor building closed cooling 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.

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the reactor building closed cooling water pumps differential pressure test, Surveillance Requirement 4.7.3.1.a.2, a substantial flow test, is to ensure that the pumps have not degraded to a point where the accident analysis

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM (Continued)

would be adversely impacted. The surveillance requirement acceptance criteria for the reactor building closed cooling water pumps was developed assuming a 7% degraded pump from the actual pump curves. Flow measurement instrument inaccuracy for the reactor building closed cooling water pumps have been accounted for in the design basis hydraulic analysis. Pressure measurement instrument inaccuracy for the reactor building closed cooling water pumps is accounted for in the acceptance criteria contained in the surveillance procedure.

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.

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the service water pumps differential pressure test, Surveillance Requirement 4.7.4.1.a.2, a substantial flow test, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the service water pumps was developed assuming a 7% degraded pump from the actual pump curves. Flow and pressure measurement instrument inaccuracies for the service water pumps have been accounted for in the design basis hydraulic analysis. It is not necessary to account for flow and pressure measurement instrument inaccuracies in the acceptance criteria contained in the surveillance procedure.

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.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

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 ± 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.

The MODES 5 and 6 action requirement to suspend positive reactivity additions does not preclude completion of actions to establish a safe conservative plant condition.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 236

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 January 4, 1999, as supplemented April 7, 1999, the Northeast Nuclear Energy Company, et al. (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 2 Technical Specifications (TS). The requested amendment would change TS 3.5.2, "Emergency Core Cooling Systems – ECCS Subsystems - Tavg ≥ 300 °F;" TS 3.6.2.1, "Containment Systems - Depressurization and Cooling Systems - Containment Spray and Cooling Systems;" TS 3.7.1.2, "Plant Systems - Auxiliary Feedwater Pumps;" TS 3.7.3.1, "Plant Systems - Reactor Building Closed Cooling Water System;" and TS 3.7.4.1, "Plant Systems - Service Water System." The Bases of the associated TS would be modified to address the proposed changes. The proposed changes to each of the surveillance requirement acceptance criteria are consistent with revised hydraulic and accident analyses performed by the licensee. Also, changes are proposed to measure pump performance in terms of differential pressure instead of discharge pressure. The April 7, 1999, supplemental letter did not change the staff's original proposed no significant hazards consideration determination.

2.0 EVALUATION

Each change proposed by the licensee is discussed in the following paragraphs along with the staff's corresponding evaluation.

2.1 TS 3.5.2

The licensee is proposing to revise surveillance requirement 4.5.2a.1.b to change the acceptance criterion from ≥1231 pounds per square inch (psi) to ≥1193 pounds per square inch differential (psid). This changes the minimum acceptable value for the recirculation flow test of the high-pressure safety injection pump (HPSI). The revised value is based on the manufacturer's curves, with a 5 percent degradation. The associated accident analyses have

also been revised to reflect a HPSI flow that assumes 5 percent degradation. The instrument inaccuracies will be accounted for in the hydraulic analysis and the surveillance procedure acceptance criteria.

The licensee's proposed revision is acceptable because the revised acceptance criterion is consistent with the value assumed in the accident analyses. Therefore, the surveillance testing will ensure that the accident analyses assumptions will continue to be met.

Additionally, the licensee is proposing to revise surveillance requirement 4.5.2a.2.b to change the acceptance criterion from ≥157 psi to ≥163 psid. This changes the minimum acceptable value for the recirculation flow test of the low-pressure safety injection pump (LPSI). The revised value is based on the manufacturer's curves and minimum recirculation flow test results that showed pump developed heads less than the manufacturer's curves. The accident analysis values are not changed by the licensee's proposal due to the delayed start of the LPSI pump during a design basis accident. The licensee has determined that pressure instrument inaccuracies are not required to be addressed since the accident analyses do not credit LPSI flow at recirculation flow conditions. Pressure measurement instrument inaccuracy will be applied and controlled by the surveillance procedures when verifying pump performance in the flow ranges credited in the accident analyses. Flow measurement instrument inaccuracy has been accounted for in the hydraulic analysis.

The licensee's proposed revision is acceptable because it provides a corrected value for the pump acceptance criterion that will allow future testing to detect pump degradation.

2.2 TS 3.6.2.1

The licensee is proposing to revise surveillance requirement 4.6.2.1.1a.2 to change the acceptance criterion from a discharge pressure of ≥254 psig (pounds per square inch gauge) to a differential pressure of ≥232 psid. This changes the minimum acceptable value for the recirculation flow test of the containment spray pumps. The revised value is based on a minimum flow curve developed from field test data, with a 5 percent degradation. This flow value is consistent with the revised hydraulic and accident analyses. Flow and pressure measurement instrument inaccuracies are already reflected in the TS acceptance criteria.

The licensee's proposed revision is acceptable because the revised acceptance criterion still bounds the design basis accident assumptions. Additionally, changing the measurement from psig to a differential pressure will compensate for possible differences in suction pressure.

2.3 TS 3.7.1.2

The licensee is proposing to revise surveillance requirement 4.7.1.2a.2.a to change the acceptance criterion from a discharge pressure of ≥1070 psig to a differential pressure of ≥1144 psid. This changes the minimum acceptable value for the recirculation flow test of the motor driven auxiliary feedwater pumps. The revised value is based on pump curves, with a 5 percent degradation. This flow value is consistent with the revised hydraulic and accident analyses. Pressure and flow measurement instrument inaccuracies will be applied to test data when verifying pump performance in the flow ranges credited in the accident analyses.

The licensee's proposed revision is acceptable because the revised acceptance criterion bounds the design basis accident assumptions. Additionally, changing the measurement from psig to a differential pressure will compensate for possible differences in suction pressure.

The licensee is proposing to revise surveillance requirement 4.7.1.2a.2.b to change the acceptance criterion from a discharge pressure of ≥1080 psig to a differential pressure of ≥1113 psid, corrected to rated pump speed. This changes the minimum acceptable value for the recirculation flow test of the turbine driven auxiliary feedwater pump. This value is different from the value proposed in the licensee's submittal of January 4, 1999.

The licensee amended their submittal in a letter dated April 7, 1999, because it was determined through testing that the pump could not meet the previously proposed criterion of ≥1134 psid. The licensee stated in this submittal that the new proposed acceptance criterion is based on high-flow test data extrapolated to minimum recirculation flow. The staff questioned this methodology because it is highly unlikely that the hydraulic performance of a centrifugal pump operating at recirculation flow rates can accurately be extrapolated from high-flow performance data unless the extrapolation is bounded by test data. In a phone conversation with the licensee on April 14, 1999, the licensee stated that they had performance data at several points in the low-flow region which differed from the original manufacturer's pump curve. In addition, the accident analysis does not credit the auxiliary feedwater pump system below flow rates of 300 gpm. The turbine driven auxiliary feedwater pump does not start automatically at Millstone 2, but is initiated manually by procedure. Therefore, there do not appear to be any concerns about this pump operating for extended periods of time at low-flow conditions other than for testing.

The licensee's proposed revision is acceptable because the revised acceptance criterion has been derived from actual test data. Additionally, changing the measurement from psig to a differential pressure will compensate for possible differences in suction pressure. The addition of the phrase "corrected to rated pump speed" allows the licensee to correct the test results for variations in the speed of the turbine driven pump during the test.

2.4 TS 3.7.3.1

The licensee is proposing to revise surveillance requirement 4.7.3.1a.2 to change the acceptance criterion from a discharge pressure verification to a differential pressure verification for the reactor building closed cooling water pumps. As with the above examples, this provides for a more meaningful measure of the pump's performance because variations in suction pressure will no longer affect the test.

2.5 TS 3.7.4.1

The licensee is proposing to revise surveillance requirement 4.7.4.1a.2 to change the acceptance criterion from a discharge pressure verification to a differential pressure verification for the service water pumps. As with the above examples, this provides for a more meaningful measure of the pump's performance because variations in suction pressure will no longer affect the test.

2.6 TS Bases Changes

The licensee also proposed changes to the corresponding TS Bases to reflect the above changes to the TS. The staff found that the licensee's TS Bases changes provided appropriate amplifying information for the plant operators.

2.7 Evaluation Summary

The licensee proposed several changes to the TS as a result of revised hydraulic analyses and related accident analyses. The licensee also changed the acceptance criteria from pump discharge pressure to pump differential pressure to eliminate the pump suction pressure variability from affecting the test. The licensee further stated that the proposed changes will ensure that pump degradation that could adversely impact the accident analyses will be detected. As discussed above, the staff reviewed the information provided by the licensee and determined that the licensee's proposed changes were 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 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 (64 FR 2523, January 14, 1999). 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.

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