50-336

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

February 8, 1999

Mr. Martin L. Bowling, Jr.
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 NO. MA2340)

Dear Mr. Bowling:

The Commission has issued the enclosed Amendment No. 226 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2, in response to your application dated July 21, 1998, as supplemented October 6, December 16, and December 31, 1998.

The amendment changes various Reactor Protection System (RPS) and Engineered Safety Feature Actuation System setpoints and allowable values; corrects the specified maximum reactor power level limited by the high power level RPS trip; adds a new Technical Specification associated with the automatic isolation of steam generator blowdown; and makes several editorial changes to correct various errors and to provide needed clarification. The amendment also makes changes to the applicable Bases pages and expands the Bases to discuss the new requirements for the automatic isolation of steam generator blowdown. However, the staff has not completed its evaluation of the requested change in the trip setpoint and allowable values for the steam generator water level. This portion of your request will be addressed later.

dfo1

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,

Stephen Dembek

Stephen Dembek, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 226 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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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,

Stephen Dembek, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures: 1. Amendment No. 226 to DPR-65
2. Safety Evaluation

cc w/encls: See next page

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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. 226
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 July 21, 1998, as supplemented October 6, December 16, and December 31, 1998, 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. 226, 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

William M. Dean

William M. Dean, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 8, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 226

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
VIII	VIII
2-2	2-2
2-4	2-4
2-5	2-5
B 2-1	B 2-1
B 2-3	B 2-3
B 2-4	B 2-4
B 2-5	B 2-5
B 2-7	B 2-7
3/4 3-4	3/4 3-4
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-24	3/4 3-24
-	3/4 7-9d
B 3/4 3-2a	B 3/4 3-2a
B 3/4 7-3a	B 3/4 7-3a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

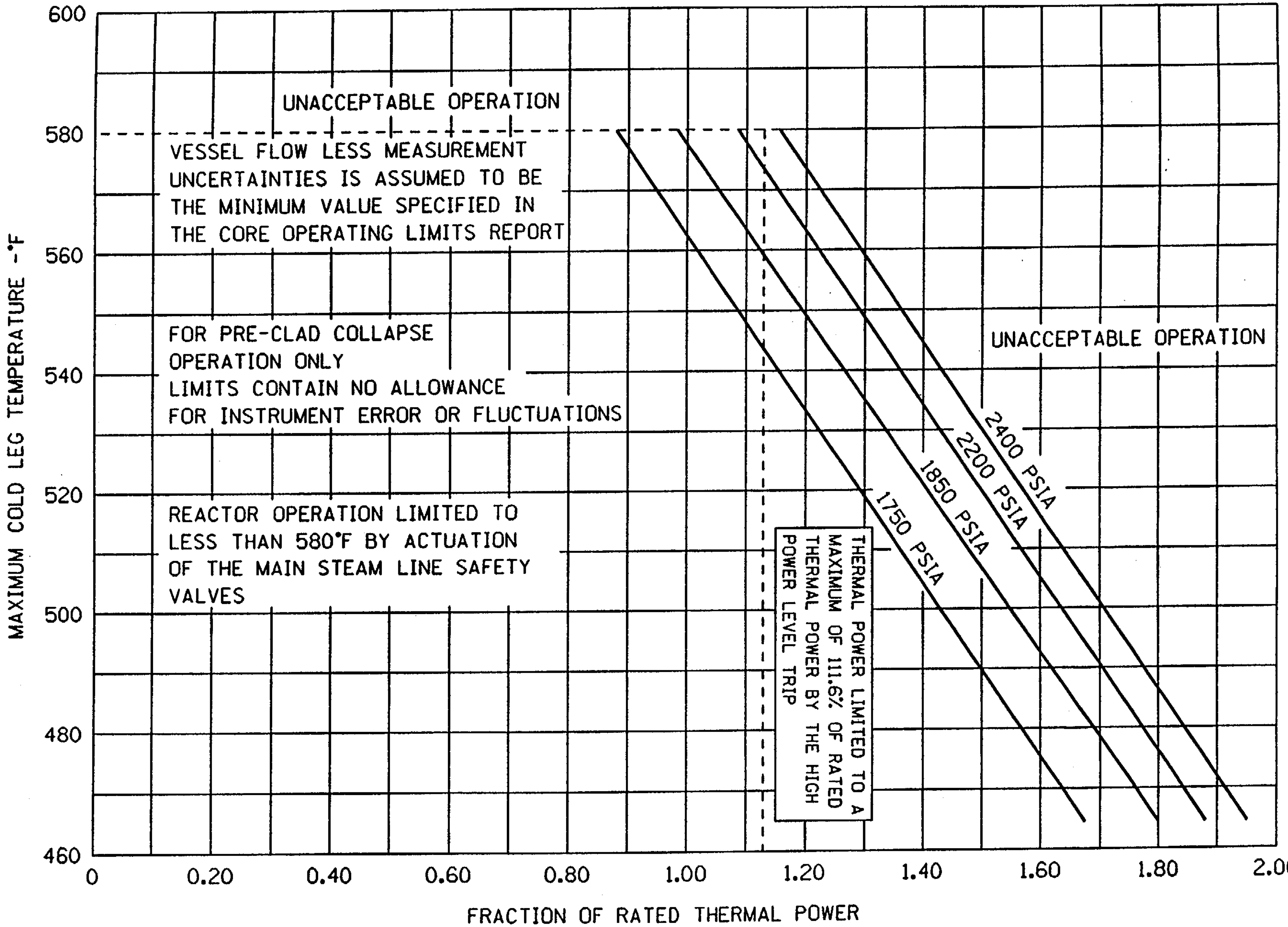

FIGURE 2.1-1
REACTOR CORE THERMAL MARGIN SAFETY LIMIT-
FOUR REACTOR COOLANT PUMPS OPERATING

SKM-MCR-21-1.dgn

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Level-High		
	Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% Above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3.	Reactor Coolant Flow - Low (1)	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.9% of reactor coolant flow with 4 pumps operating.
4.	Reactor Coolant Pump Speed - Low (1)	≥ 830 rpm	≥ 823 rpm
5.	Pressurizer Pressure - High	≤ 2397 psia	≤ 2407 psia
6.	Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
7.	Steam Generator Pressure - Low (2) (5)	≥ 691 psia	≥ 677 psia
8.	Steam Generator Water Level - Low (5)	≥ 36.0% Water Level - each steam generator	≥ 35.2% Water Level - each steam generator
9.	Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

*Design Reactor Coolant flow with 4 pumps operating is the lesser of either:
a. The reactor coolant flow rate measured per Specification 4.2.6.1, or
b. The minimum value specified in the CORE OPERATING LIMITS REPORT.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).
11. Loss of Turbine--Hydraulic Fluid (3) Pressure - Low	$\geq$ 500 psig	$\geq$ 500 psig

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is $\geq$ 800 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Calculations of the trip setpoint includes measurements, calculational and processor uncertainties, and dynamic allowances.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the XNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.17. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.17. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperatures is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 111.6% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in figure 2.1-1 to be valid are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.17 and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SET POINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.6% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 106.6% of RATED THERMAL POWER and a minimum setpoint of 14.6% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 111.6% of RATED THERMAL POWER, which is the value used in the accident analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protection system to permit

Reactor Coolant Flow-Low (Continued)

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. However, power operation with fewer than four reactor coolant pumps operating has not been analyzed and is prohibited. The low-flow trip setpoint and Allowable Value have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.17 under normal operation and expected transients.

Pressurizer Pressure-High

The pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is approximately 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpont for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The trip setting is sufficiently below the full-load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow.

Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1850 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2.25°F to compensate for potential temperature measurement uncertainty; and a further allowance of 74 psi to compensate for pressure measurement error, trip system processing error, and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 74 psi allowance is made up of a 5 psi bias, a 19 psi pressure measurement allowance and a 50 psi time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuring transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is $\geq$ 800 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip does not need to be operable if all the control rod drive mechanisms are de-energized or if the RCS boron concentration is greater than or equal to the refueling concentration of Specification 3.9.1.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) ΔT Power input to trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, within 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. AUXILIARY FEEDWATER					
a. Manual	1/pump	1/pump	1/pump	1, 2, 3	1
b. Steam Generator Level - Low	4	2	3	1, 2, 3	2
10. STEAM GENERATOR BLOWDOWN					
a. Steam Generator Level - Low	4	2	3	1, 2, 3	2

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed when pressurizer pressure is < 1850 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1850 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed when steam generator pressure is < 700 psia; bypass shall be automatically removed when steam generator pressure is $\geq$ 700 psia.

(d) Deleted

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, within 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ACTION 3 - With less than the minimum channels OPERABLE the containment purge valves are to be maintained closed.

ACTION 4 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1850 psia: immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing the pressurizer pressure above 1850 psia.

b. $\geq$ 1850 psia, operation may continue with the inoperable channel in the bypassed condition, provided the following condition is satisfied:

1. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.2.1.1 provided BOTH of the inoperable channels are placed in the bypassed condition.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 9.48 psig	≤ 10.11 psig
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Steam Generator Pressure - Low	≥ 572 psia	≥ 558 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. ENCLOSURE BUILDING FILTRATION (EBFAS)		
a. Manual EBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank -Low	46 ± 3 inches above tank bottom	46 ± 6 inches above tank bottom
7. CONTAINMENT PURGE VALVES ISOLATION		
a. Containment Radiation - High		
Gaseous Activity	≤ the value determined in accordance with Specification 4.3.2.1.4.	≤ the value determined in accordance with Specification 4.3.2.1.4.
Particulate Activity (Half Lives greater than 8 days)	≤ the value determined in accordance with Specification 4.3.2.1.4.	≤ the value determined in accordance with Specification 4.3.2.1.4.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level one	≥ 2912 volts with a 2.0 ± 0.1 second time delay	≥ 2877 volts with a 2.0 ± 0.1 second time delay
b. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level two	≥ 3700 volts with an 8.0 ± 2.0 second time delay	≥ 3663 volts with an 8.0 ± 2.0 second time delay
9. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%
10. STEAM GENERATOR BLOWDOWN		
a. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%

PLANT SYSTEMS

STEAM GENERATOR BLOWDOWN ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.8 Each steam generator blowdown isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With one or more steam generator blowdown isolation valves inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours; or

b. Isolate the affected steam generator blowdown line within 4 hours; or

c. Be in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.8 Verify the closure time of each steam generator blowdown isolation valve is $\leq$ 10 seconds on an actual or simulated closure signal at least once per 18 months.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. CONTAINMENT SUMP RECIRCULATION (SRAS)				
a. Manual SRAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
7. CONTAINMENT PURGE VALVES ISOLATION				
a. Containment Radiation - High				
Gaseous Monitor	S	R	M	ALL MODES
Particulate Monitor	S	R	M	ALL MODES
8. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level one	S	R	M	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level two	S	R	M	1, 2, 3
9. AUXILIARY FEEDWATER				
a. Manual	N.A.	N.A.	R	N.A.
b. Steam Generator Level - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M	1, 2, 3
10. STEAM GENERATOR BLOWDOWN				
a. Steam Generator Level - Low	S	R	M	1, 2, 3

Steam Generator Blowdown Isolation

Automatic isolation of steam generator blowdown will occur 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.

Sensor Cabinet Power Supply Auctioneering

The auctioneering circuit of the ESFAS sensor cabinets ensures that two sensor cabinets do not de-energize upon loss of a D.C. bus, thereby resulting in the false generation of an SRAS. Power source VA-10 provides normal power to sensor cabinet A and backup power to sensor cabinet D. VA-40 provides normal power to sensor cabinet D and backup power to cabinet A. Power sources VA-20 and VA-30 and sensor cabinets B and C are similarly arranged.

If the normal or backup power source for an ESFAS Sensor Cabinet is lost, two sensor cabinets would be supplied from the same power source, but would still be operating with no subsequent trip signals present. However, any additional failure associated with this power source would result in the loss of the two sensor cabinets, consequently generating a false SRAS. The 48-hour Action Statement ensures that the probability of a Action Statement and an additional failure of the remaining power source, while in this Action Statement is sufficiently small.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The spent fuel storage area monitors provide a signal to direct the ventilation exhaust from the spent fuel storage area through a filter train when the dose rate exceeds the setpoint. The filter train is provided to reduce the particulate and iodine radioactivity released to the atmosphere. Should an accident involving spent fuel occur, the 100 mR/hr actuation setpoint would be sufficient to limit any consequences at the exclusion area boundary to those evaluated in the NRC Safety Evaluation, Section 15 (May 1974).

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226

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 July 21, 1998, as supplemented October 6, December 16, and December 31, 1998, 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 proposed amendment would change various Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) setpoints and allowable values; correct the specified maximum reactor power level limited by the high power level RPS trip; add a new TS associated with the automatic isolation of steam generator (SG) blowdown; and make several editorial changes to correct various errors and to provide needed clarification. The amendment would also make changes to the applicable Bases pages and expand the Bases to discuss the new requirements for the automatic isolation of SG blowdown. The October 6, December 16, and December 31, 1998, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The proposed revision to the TS will change trip setpoints (SPs) and allowable values (AVs) of the RPS and ESFAS instrumentation. In addition, a new TS requirement for operability and closure time of the SG blowdown isolation valves (TS 3.7.1.8) will be added in addition to changes to the associated Bases sections and other minor miscellaneous changes including editorial changes. TS sections affected are TS 2.1.1, "Safety Limits - Reactor Core"; TS 2.2.1, "Limiting Safety System Settings - Reactor Trip Set Points"; TS 3.3.1.1, "Instrumentation - Reactor Protective Instrumentation"; TS 3.3.2.1, "Instrumentation - Engineered Safety Feature Actuation System Instrumentation"; and the Bases sections associated with the revised TS sections.

The licensee recently revised the instrument uncertainty and setpoint calculations associated with the RPS and the ESFAS instrumentation to address the effects of harsh environment parameters (pressure, temperature, and radiation). The proposed changes to the SPs and AVs are the result of revisions to these calculations. The licensee's current analysis for loss of main feed water assumes an automatic isolation of SG blowdown isolation valves. Therefore, it is necessary to add a new TS to reflect this analysis assumption.

3.0 PROPOSED CHANGES AND EVALUATION

3.1 RPS and ESFAS SP Changes

3.1.1 Proposed Changes

(a) TS Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits"

- Pressurizer Pressure - High: Revise Trip SP from "≤ 2400 psia" to "≤ 2397 psia," and the AV from "≤ 2408 psia" to "≤ 2407 psia."

- Containment Pressure - High: Revise the Trip SP from "≤ 4.75 psig" to "≤ 4.42 psig," and the AV from "≤ 5.24 psig" to "≤ 5.07 psig."

- Steam Generator Pressure - Low: Revise Trip SP from "≥ 680 psia" to "≥ 691 psia," and the AV from "≥ 672 psia" to "≥ 677 psia."

- Steam Generator Water Level - Low: Revise Trip SP from "≥ 36.0%" to "≥ 39.1%," and the AV from "≥ 35.2% "to "≥ 38.0 %."

- Revise footnote (2) to replace the words, "below 780 psia when" with the words "when steam generator pressure is ‹ 800 psia and," and the words "at or above 780 psia," with the words, "when steam generator pressure is ≥ 800 psia."

(b) TS Table 3.3-1, "Reactor Protective Instrumentation"

- Revise footnote (b) to replace the words, "below 780 psia when" with the words "when steam generator pressure is ‹ 800 psia and," and the words "at or above 780 psia," with the words, "when steam generator pressure is ≥ 800 psia."

(c) TS Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation"

- Revise footnote (a) to change the pressurizer pressure value from "1750" psia to "1850" psia.

- Revise footnote (c) to replace the words, "below 600 psia," with the words "when steam generator pressure is ‹ 700 psia," and the words "at or above 600 psia," with the words, "when steam generator pressure is ≥ 700 psia."

- Revise a. and b. of ACTION 2 and ACTION 4 to change the value of pressurizer pressure from "1750" psia to "1850" psia.

(d) TS Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values"

- Safety Injection (SIAS), Containment Pressure - High; Containment Isolation (CIAS), Containment Pressure - High; Main Steam Line Isolation, Containment Pressure - High; and Enclosure Building Filtration (EBFAS), Containment Pressure - High: Revise the Trip SP from "≤ 4.75 psig" to "≤ 4.42 psig," and the AV from "≤ 5.20 psig" to "≤ 5.07 psig."

- Safety Injection (SIAS), Pressurizer Pressure - Low; Containment Isolation (CIAS), Pressurizer Pressure - Low; and Enclosure Building Filtration (EBFAS), Pressurizer Pressure - Low: Revise the Trip SP from "≥1600 psia" to "≥ 1714 psia," and the AV from "≥ 1592.5 psia" to "≥ 1704 psia."

- Main Steam Line Isolation, Steam Generator Pressure - Low: Revise the Trip SP from "≥ 500 psia" to "≥ 572 psia," and the AV from "≥ 492.5 psia" to "≥ 558 psia."

- Containment Sump Recirculation (SRAS), Refueling Water Storage Tank - Low: Revise the Trip SP from "48 ± 9" inches to "46 ± 3" inches above tank bottom, and the AV from "48 ± 18" inches to "46 ± 6" inches above tank bottom.

- Auxiliary Feedwater, Steam Generator Level - Low: Revise the Trip SP from ≥ "12%" to ≥ "26.8%," and the AV from ≥ "10%" to ≥ "25.2%."

3.1.2 Evaluation

The licensee revised the existing SP calculations to include changes to the ALs and to address the effects of a harsh environment (pressure, temperature, and radiation) on instrument SP uncertainties appropriately. In its submittal, the licensee stated that the previous SP calculations were performed using guidance provided in industry standard, ISA-RP67.04, Part II, 1994, "Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation." This standard provides several methods for performing SP calculations. The licensee stated that its recent reevaluation of the SP calculation methodology determined that using another method (Method #1) within ISA-RP67.04, Part II, 1994, instead of the original method (Method #2), would provide a larger difference between the trip SP and its AV for the same total instrument loop uncertainty, and therefore, will result in a more conservative trip SP. Therefore, the licensee decided to use Method #1 for performing the revised SP calculations.

The staff has not yet approved ISA-RP67.04, Part II, 1994, for generic use in SP calculations. The approved standard, which provides guidance for SP calculations, is ISA 67-04, 1982, which is endorsed by the staff in Regulatory Guide (RG) 1.105, Revision 2. In its submittal, the licensee stated that the revised SP calculations use a 24-month refueling cycle instrument drift value instead of the instrument drift associated with the normal surveillance frequency of

18 months, even though the licensee has not requested approval to revise its TS surveillance frequency from the existing 18 months to 24 months. The licensee stated that the use of a longer refueling cycle for instrument drift values to locate the nominal SP is more conservative.

Since the licensee is not using the generically approved calculation method for SP/AV calculations, the staff, by a letter dated September 16, 1998, requested that the licensee provide additional information relating to its method for treatment of uncertainty elements, calculation of nominal SP, AV, and their relation to each other and to the AL, location of the nominal SP, evaluation of drift, SP assumptions, and verification of the SP through routine surveillance tests and test acceptance criteria for such tests. In response to the staff's request for information, the licensee, by letter dated October 6, 1998, provided the following explanation:

- Millstone Unit 2 is not committed to either RG 1.105, Revision 2, or ISA 67.04, 1982, but the treatment of elements of instrument loop uncertainty in the Millstone Unit 2 methodology is consistent with the guidance in RG 1.105, Revision 2, and ISA 67.04, 1982.

- The methodology ensures that the selected nominal SPs will initiate mitigating action(s) prior to the monitored variable encroaching on its AL by appropriately considering all elements of instrument uncertainty and response time of the actuated equipment.

- The SP methodology requires the preparer of SP calculations to determine the scope of each channel tested in the monthly and refueling procedures. The allowances used to calculate the SP and AV are then based on the surveillance procedures. The SP methodology also includes the calculation of as-found and as-left acceptance criteria for the monthly and refueling interval surveillance procedures. The acceptance criteria in the surveillance procedures must be at least as restrictive as the values determined in the SP calculations. Engineering review of surveillance procedures ensures the correct implementation of the trip SP, AV, and test acceptance criteria.

- The historical data and the drift analysis indicated that in many cases the drift did not increase with time and 22.5 months drift (18 + 25%) was equal to 30 months (24 + 25%) drift. In a few cases where drift was found to be time dependent, a conservative multiplier was used to determine bounding drift for 30 months.

The staff has not completed its evaluation of the specific SP and AV for the Steam Generator Water Level-Low. The acceptability of the proposed SP and AV for this parameter will be addressed later.

Although the licensee did not use the staff's generically approved method for SP/AV calculations, based on its review of the licensee's response to the staff's request for information, the staff concludes that the licensee's calculation method meets the intent of RG 1.105, Revision 2, and ISA 67.04, 1982. In addition, the staff has reviewed the changes to the footnotes and Action 4 and has concluded that the RPS and ESFAS will continue to function as before. The change to Action 2 requested by the licensee is no longer applicable due to changes made to the TS in License Amendment No. 225, dated January 27, 1999. Therefore, the staff finds the TS changes, as previously described, to be acceptable.

3.2 New TS Requirements for SG Blowdown Isolation

3.2.1 Proposed Changes

- TS Index Page VIII: Revise page to reflect the addition of TS 3.7.1.8, "Steam Generator Blowdown Isolation Valves."

- TS Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation." Add a new Functional Unit, 10.a, Steam Generator Blowdown - Steam Generator Level - Low, with the column for "Total No. of Channels" designated as "4," with the column for "Channels to Trip" designated as "2," with the column for "Minimum Channels Operable" designated as "3," with the column for "Applicable Modes" designated as "1, 2, 3," and with the column for "Action" designated as "2."

- TS Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values." Add a new Functional Unit, 10.a, "Steam Generator Blowdown - Steam Generator Level - Low," with a Trip SP of "≥ 26.8%" and the AV of "≥ 25.2%."

- TS Table 4.3-2, "Engineered Safety Features Actuation System Instrumentation Surveillance Requirements." Add a new Functional Unit 10.a, "Steam Generator Blowdown - Steam Generator Level - Low," with the column for "Channel Check" designated as "S," with the column for "Channel Calibration" designated as "R," with the column for "Channel Functional Test" designated as "M," and with the column for "Modes in Which Surveillance Required" designated as "1, 2, 3."

- TS 3.7.1.8, "Steam Generator Blowdown Isolation Valves." Add a new TS for the SG blowdown isolation valves. The new TS would include the limiting condition for operation, applicability, action, and surveillance requirements.

3.2.2 Evaluation

In its submittal, the licensee stated that the current analysis for the loss of main feedwater assumes an automatic isolation of SG blowdown. Therefore, the licensee stated that the operability of the SG blowdown isolation valves should be required by the TS. Therefore, the licensee proposed adding requirements to the TS to support operability verification of the SG blowdown isolation valves. The licensee stated that the proposed TS requirements are consistent with other engineered safety feature functions. The licensee stated that isolation of SG blowdown will occur at the same SG level SP as is used for actuation of the auxiliary feedwater (AFW) system. The licensee further stated that the SG blowdown isolation valves are also containment isolation valves; therefore, the proposed action and surveillance requirements are consistent with TS 3.6.3.1, "Containment Isolation Valves."

The licensee stated that the proposed change to add the automatic isolation of SG blowdown on low SG level will require modification of the AFW actuation and SG blowdown isolation circuits. The components added will be seismically supported and qualified as Class 1E. They have the same reliability as the existing seismically supported, Class 1E components in the AFW control system and blowdown isolation circuits. The added logic is redundant so no single failure can cause the logic to fail to function and the mechanical failure of a SG blowdown isolation valve does not represent a more limiting single failure than is currently considered in the transient analysis. The added logic is in the form of an "or" logic with the existing system logic so no operation or failure of the added feature can prevent the existing logic from performing its intended function.

Therefore, based on the preceding evaluation, the staff finds the proposed surveillance requirements and associated proposed actions for the SG blowdown isolation valves acceptable.

3.3 Miscellaneous Changes

3.3.1 Proposed Change

- TS Figure 2.1-1, "Reactor Core Thermal Margin Safety Limit - Four Reactor Coolant Pumps Operating." Revise the value of the maximum thermal power that is limited by the high power level trip, from "112%" to "111.6%."

3.3.2 Evaluation

In its submittal, the licensee stated that this value is currently used in the safety analysis and is consistent with the maximum allowed high power trip SP of 106.6% + 5% uncertainty, which was approved by the staff in License Amendment No. 61, dated October 6, 1980. The staff has reviewed the change and finds it acceptable.

3.4 Editorial Changes

3.4.1 Proposed Changes

(a) Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits"

- In footnote (3), replace the word "IS" by the word "is."

- In footnote (5), replace the word "steram" by the word "steam."

(b) TS Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values"

- Add "≤" to the trip SP in Functional Unit 1.b.

- Replace the word 'VALUE" with "SETPOINT" for the heading of the second column on page 3/4 3-19.

- Add "Amendment No. ~~45~~, ~~48~~, ~~72~~" to the bottom of page 3/4 3-19 to show that the page was previously changed by license amendment numbers 45, 48, and 72.

3.4.2 Evaluation

The above proposed changes are not technical changes but are editorial only. As stated by the licensee in its submittal, implementation of these changes will improve the clarity of the statement(s). Therefore, the proposed changes are acceptable to the staff.

3.5 Changes to Bases Sections

The staff has reviewed the changes to the applicable TS Bases sections and the expansion of the Bases to discuss the new requirements for the automatic isolation of steam generator blowdown. Also, Bases page B 3/4 7-3a, was changed to correct an error that was introduced in License Amendment No. 223, dated December 31, 1998. The staff finds the changes appropriate.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement 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 43208 dated August 12, 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.

6.0 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.

Principal Contributor: S. Athavale

Date: February 8, 1999