August 10, 1992

Docket No. 50-443

Mr. Ted C. Feigenbaum, Senior Vice President
and Chief Nuclear Officer
North Atlantic Energy Service Corporation
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: ISSUANCE OF AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. NPF-86 - SEABROOK STATION, UNIT NO. 1 (TAC NO. M83079)

The Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. NPF-86 for the Seabrook Station. This amendment is in response to your application dated March 20, 1992, as supplemented on June 19, 1992.

This amendment provides for replacement of the Resistance Temperature Detector (RTD) Bypass System for measurement of primary loop temperature with fast-response thermowell-mounted RTDs. Additionally, the changes include specification of a thermal design flow plus a flow measurement uncertainty for primary loop piping, and permit a precision heat balance to be performed above 95 percent of full power instead of 75 percent of full power.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER notice.

Sincerely,

/S/
Gordon Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 12 to License No. NPF-86
2. Safety Evaluation

cc w/enclosures: See next page

LA:PDI-3	PE:PDI-3	PM:PDI-3	OGC w/note	(A)D:PDI-3	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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G E Edison

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Division of Reactor Projects - I/II
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2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. NPF-86

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

   A. The application for amendment filed by the North Atlantic Energy Service Corporation (NAESCO) (the licensee), acting for itself and as agent and representative of the 11 other utilities listed below and hereafter referred to as licensees, dated March 20, 1992, as supplemented on June 19, 1992, 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 set forth in 10 CFR Chapter I;

   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.

*North Atlantic Energy Service Corporation is authorized to act as agent for the North Atlantic Energy Corporation, the Canal Electric Company, The Connecticut Light and Power Company, EUA Power Corporation, Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, New Hampshire Electric Cooperative, Inc., Taunton Municipal Light Plant, The United Illuminating Company, and Vermont Electric Generation and Transmission Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-86 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 12 , and the Environmental Protection Plan contained in Appendix B are incorporated into Facility License No. NPF-86. NAESCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance. The upgrades and enhancements associated with replacement of the RTD Bypass System in this amendment will be implemented prior to entry into Mode 3 during restart from the second refueling outage. The resistance thermal detector cross-calibration and response time tests, and a reactor coolant system leak test can be completed following entry into Mode 3. Additionally, a flow calorimetric measurement will be performed upon achieving stable full power operation. All other testing required to demonstrate proper operation of modified components will be completed prior to entry into Mode 3.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses

Victor Nerses, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 10, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

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 area of change. Overlap pages have been provided.

Remove	Insert
2-4	2-4
2-5	2-5
2-7	2-7
2-8	2-8
2-10	2-10
3/4 2-10	3/4 2-10
3/4 3-9	3/4 3-9
3/4 3-13	3/4 3-13
B 3/4 2-4	B 3/4 2-4

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 2.2-1 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤31.1% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6×10^5 cps
7. Overtemperature ΔT	6.5	3.5	1.7** +0.5**	See Note 1	See Note 2
8. Overpower ΔT	4.9	2.2	1.7	See Note 3	See Note 4
9. Pressurizer Pressure - Low	3.12	0.86	0.99	≥1945 psig	≥1,931 psig
10. Pressurizer Pressure - High	3.12	1.00	0.99	≤2385 psig	≤2,398 psig

*RTP = RATED THERMAL POWER

**The sensor error for T_{avg} is 1.7 and the sensor error for Pressurizer Pressure is 0.5. "As measured" sensor errors may be used in lieu of either or both of these values, which then must be summed to determine the overtemperature ΔT total channel value for S.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	2.5	1.9	0.6	≥90% of loop design flow*	≥89.3% of loop design flow*
13. Steam Generator Water Level Low - Low	14.0	12.53	0.55	≥14.0% of narrow range instrument span	≥12.6% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*Loop design flow = 95,700 gpm

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	$\leq$10% of RTP*	$\leq$12.1% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	$\leq$10% RTP* Turbine Impulse Pressure Equivalent	$\leq$12.3% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	$\leq$50% of RTP*	$\leq$52.1% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	$\leq$20% of RTP*	$\leq$22.1% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	$\geq$10% of RTP*	$\geq$7.9% of RTP*
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	$\leq$10% RTP* Turbine Impulse Pressure Equivalent	$\leq$12.3% RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_o \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \frac{(1)}{(1 + \tau_6 S)} - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where:

ΔT = Measured ΔT by RTD Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_1 = 1.0995;

K_2 = 0.0112/°F;

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 33$ s, $\tau_5 \leq 4$ s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

T' $\leq$ 588.5°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.000519/psig;

P = Pressurizer pressure, psig;

P' = 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests so that:

(1) For $q_t - q_b$ between - 35% and + 8%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 35%, the ΔT Trip Setpoint shall be automatically reduced by 1.09% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 8%, the ΔT Trip Setpoint shall be automatically reduced by 1.00% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_o \{K_4 - K_5 \frac{(\tau_7 S)}{(1 + \tau_7 S)} \frac{(1)}{(1 + \tau_6 S)} T - K_6 [T \frac{(1)}{(1 + \tau_6 S)} - T''] - f_2(\Delta I)\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	=	1.09,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.001386/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,

T = As defined in Note 1,

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F),

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of ΔT span.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ (by F_{xy} evaluation) and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric thimble locations or

b. Using the movable incore detection system to monitor the QUADRANT POWER TILT RATIO subject to the requirements of Specification 3.3.3.2.

*See Special Test Exceptions Specification 3.10.2.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the the following limits:

a. Reactor Coolant System T_{avg}, $\leq$ 594.3°F

b. Pressurizer Pressure, $\geq$ 2205 psig*

c. Reactor Coolant System Flow, $\geq$ 392,000 gpm**

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a precision heat balance measurement to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Includes a 2.4% flow measurement uncertainty.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(13)	N.A.	1, 2, 3*,
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q(16)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(16)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(16)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),Q(9,16)	N.A.	N.A.	2**, 3, 4,
7. Overtemperature ΔT	S	R	Q(16)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q(16)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(16,17)	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	Q(16,17)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q(16)	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R	Q(16)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-- Low-Low	S	R	Q(16, 17)	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q(16)	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q(16)	N.A.	1
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(12) Number not used.

(13) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(14) Local manual shunt trip prior to placing breaker in service.

(15) Automatic undervoltage trip.

(16) Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.

(17) These channels also provide inputs to ESFAS. Comply with the applicable MODES and surveillance frequencies of Specification 4.3.2.1 for any portion of the channel required to be OPERABLE by Specification 3.3.2.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 3.3-4 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

Fuel rod bowing reduces the value of DNBR. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of 1.30 vs. 1.28,

b. Grid spacing (K_s) of 0.046 vs. 0.059,

c. Thermal diffusion coefficient of 0.038 vs. 0.059,

d. DNBR multiplier of 0.86 vs. 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) as provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit of a measurement error of 4% for $F^N_{\Delta H}$ has been allowed for in determination of the design DNBR value.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly incore flux maps. During normal operation the QUADRANT POWER TILT RATIO is set equal to zero once acceptability of core peaking factors has been established by review of incore maps. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits of 594.3°F for T_{avg} and 2205 psig for pressurizer are not exceeded.

The measurement error of 2.4% for RCS total flow rate is based upon performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is applied. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. NPF-86

NORTH ATLANTIC ENERGY SERVICE CORPORATION

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated March 20, 1992, as supplemented on June 19, 1992, the Public Service Company of New Hampshire (former licensee) submitted a request for changes to the Seabrook Station, Technical Specifications (TS). Pursuant to an order authorizing transfer of the facility, North Atlantic Energy Service Corporation is now the licensed operator of Seabrook. The June 19, 1992, letter provided clarifying information in response to NRC staff's request for additional information and was renoticed in the Federal Register on July 8, 1992 (57 FR 30256) with a new evaluation of no significant hazards considerations.

The requested changes would eliminate the Resistance Temperature Detection (RTD) Bypass Manifold System, which is currently used for the measurement of narrow range Reactor Coolant System hot leg and cold leg temperature, and replace it with direct immersion RTDs. This modification affects the reactor protection system setpoints and uncertainties for RCS flow and T-average because of the different response time characteristics and instrumentation uncertainties associated with the new thermowell mounted RTDs. The T-Average and Delta-T signal input arrangement to the reactor protection and control system is also modified. Accordingly, this amendment requires a revision of the Seabrook Station Technical Specifications for Overtemperature Delta T, Overpower Delta T, Reactor Coolant Flow, and departure from nucleate boiling (DNB) parameters.

The requirements for verification of RTD bypass loop flow are deleted. The requirements for the performance of a precision heat balance calculation for determining the Reactor Coolant System flow rate are modified by increasing the thermal power level at which the heat balance is required. The submittal proposed to change the power level below which the heat balance must be done from the current requirement of 75% of rated thermal power to 95% of rated thermal power, consistent with the Westinghouse recommendation to perform the heat balance above 90% of rated thermal power.

2.0 EVALUATION

2.1 Instrumentation and Control Issues

Instrumentation and control issues are reviewed in Section 2.1 below, and rector systems issues are reviewed in Section 2.2

2.1 Current System

The present reactor coolant temperature measurement system uses coolant scoops in the primary coolant to divert a portion of the reactor coolant into bypass loops. The RTDs for T-hot and T-cold temperature measurement are located in the bypass loop manifolds and are inserted directly into the reactor coolant bypass flow without thermowells. Separate hot leg and cold leg bypass loops are provided for each reactor coolant loop such that individual T-hot and T-cold loop temperature signals can be developed for use by the reactor protection and plant control system.

Bypass piping form the hot leg side of each steam generator is used for the T-hot RTDs. Additional bypass piping from the cold leg side of the reactor coolant pump is used for the T-cold RTD. Both T-hot and T-cold manifolds empty through a common header to the intermediate leg between the steam generator and reactor coolant pump. Flow for each T-hot bypass loop is provided by three coolant scoops located at 120 degree intervals around the hot leg piping. Because temperature streaming in the cold leg is limited by the mixing action of the reactor coolant pump only one scoop connection is installed for bypass flow to the T-cold bypass manifold.

The bypass manifold system was designed to resolve concerns with temperature streaming (temperature gradients) within the hot leg primary coolant. The temperature streaming experienced in the hot leg piping is a result of the reactor coolant leaving various regions of the reactor core at different temperatures. The bypass manifold system compensates for the temperature streaming by mixing the primary coolant within the bypass manifold. The bypass manifold system also limits high velocity coolant flow to the RTDs and allows RTD replacement without the need to draindown the reactor coolant system.

The output from the bypass loop RTDs provides the signals necessary to calculate the arithmetic average loop temperature (T-average) and the loop differential temperature (Delta-T). The T-average and Delta-T signals are then input to the reactor protection system. The T-average and Delta-T signals for the plant control/computer systems are derived from the same set of protection system RTDs and T-average and Delta-T calculations. The T-average and Delta-T values are provided to the plant control/computer systems through isolation devices.

The licensee states that the current system has caused plant shutdowns due to primary leakage through valves or flanges, and the interruption of bypass flow due to valve stem failure. Additionally, the licensee stated that the bypass piping contributes to increased radiation exposure to personnel when maintenance is performed in bypass manifold system areas.

2.1.2 Proposed System

The modified system hot leg temperature measurement for each loop will be obtained using three fast response, narrow range, dual element RTDs mounted in thermowells. One element of each RTD will be utilized as a spare. Two of the hot leg RTDs will be mounted in thermowells within the existing bypass manifold scoop penetrations. Each bypass scoop will be modified such that reactor coolant will flow in through the existing holes of the bypass scoop past the RTD/thermowell assembly and out through a new hole machined in the bypass scoop. Because of structural interference a new penetration will be installed to accommodate the third RTD/thermowell assembly. The modified RTD arrangement will perform the same sampling/temperature averaging function as the original bypass manifold system. The modified location for the third RTD in each loop has been evaluated by the licensee and the revised streaming uncertainties applied to the setpoint calculations.

The cold leg temperature measurements will be obtained by one fast response, narrow range, dual element RTD located at the discharge of the reactor coolant pump. This RTD will be mounted in a thermowell within the existing cold leg bypass manifold penetration. Because of the mixing action of the reactor coolant pump, temperature gradients in the cold leg are minimized and only one RTD is used for cold leg temperature measurement. Although cold leg streaming is minimized by RCP mixing a cold leg streaming bias is incorporated into the uncertainty calculations. As in the hot leg, the bypass manifold penetration will be modified to accept the RTD thermowell.

The licensee will replace the bypass manifold direct immersion RTDs with Weed Instrument Company Inc. dual element RTDs mounted in thermowells. The spare element of each RTD will be terminated at the 7300 rack input terminals in the control room. This arrangement is intended to allow on-line accessibility to the RTD spare elements in the event of an RTD failure.

The licensee states that the new thermowell mounted RTDs have a response time equal to the time of the old bypass piping transport, thermal lag and direct immersion RTDs (about 4 seconds). The 4-second response time of the Weed RTD thermowell assembly is supported by industry experience. The 2-second electronics delay specified by the licensee is identical to the value for the RTD bypass system. The licensee concluded that the safety analysis value of 6-seconds remains valid noting that the 2-second electronic delay is conservative and provides some margin. The RTD manufacturer will perform response time testing of each RTD and thermowell prior to installation to ensure the RTD/thermowell response time is bounded by the safety analysis value. The licensee will also verify the response time of the new RTDs using loop current step response (LCSR) methodology following installation in the plant.

To accomplish the hot leg temperature function previously done by the bypass manifold system, the modified hot leg RTD temperature signals (three per loop) will be electronically averaged in the protection system. The averaged T-hot signal will then be used with the T-cold signal to calculate reactor coolant system loop Delta-T and T-average values for use in the reactor protection and plant control systems. The averaging function will be accomplished by additions to existing 7300 reactor protection equipment.

The control system T-average and Delta-T signals are derived from the reactor protection system T-average and Delta-T calculations and provided to the plant control system through isolation devices. The isolation devices and control system input methodology for T-average and ΔT are not revised per this TS amendment and continue to meet the licensee basis as outlined in Chapter 7 of the Seabrook Station Final Safety Analysis Report.

The licensee states that existing control board indicators and alarms provide a means to identify RTD failures. A cold leg RTD failure can be handled by disconnecting the failed element and connecting the spare element provided within each RCS loop.

A failure of a hot leg RTD can be managed in one of two ways. The first method disconnects the failed hot leg RTD element and reconnects the spare element of the same RTD. The second method requires plant personnel to manually defeat the failed hot leg RTD signal and rescale the electronics to average the remaining two RTD inputs. A bias value is incorporated into the T-hot average signal to compensate for hot leg streaming and maintain a value comparable with the previous three RTD average. The bias value is developed per procedure/TS requirements using data recorded at full power and during protection system surveillance.

The proposed TS changes also include a revision to the precision heat balance requirements. The licensee has modified the thermal power level at which the precision heat balance must be performed. Previously, the heat balance was performed prior to exceeding 75% of rated power. Now it will be performed prior to exceeding 95% of rated thermal power. As stated by the licensee, this is consistent with the Westinghouse recommendation to perform the precision heat balance above 90% of rated thermal power to minimize measurement uncertainties aggravated at lower power levels.

The licensee stated that following the initial thermowell RTD cross calibration, the calibration reference will consist of the average of the RTD temperatures. The staff is concerned that the use of an average RTD value as a reference during cross calibration instead of a calibrated reference may lead to a net drift of the average temperature value indicated by the RTDs over time should the installed RTDs drift systematically. The licensee indicated that RTD drift is random and with a total uncertainty of less than ± 1.2 degrees specified in the submittal. NUREG/CR-5560, "Aging of Nuclear Plant Resistance Temperature Detectors" recognizes that on-line cross calibration can be a reasonable method for RTD calibration. However, as stated in NUREG/CR-5560, to perform in-situ calibration would normally require one or more newly calibrated RTDs to be used as a reference. Without a reference the cross calibration will not account for common mode (systematic) drift and will only provide information on the consistency and not the accuracy of the installed RTDs. The cross calibration technique assumes that the average of the RTD measurements represents the true process temperature and that RTD drift is random and not systematic. The project results referenced in NUREG/CR-5560 indicate that RTD drift is usually random. However, the particular testing done to validate the cross calibration methodology in NUREG/CR-5560 utilized newly calibrated RTDs for the test.

The staff agreed with the licensee's justification for RTD calibration without a reference but will continue to evaluate cross calibration techniques on a generic basis. This is acceptable in that the bypass elimination RTDs are newly calibrated and should not be influenced by systematic drift components during the initial plant cross calibration at Seabrook.

2.1.3 Technical Specification Changes

As a result of the modifications associated with the removal of the RTD bypass manifold system, the licensee proposed various changes to the Seabrook Nuclear Station TS. The staff finds the following changes discussed in Section 2.1.3.1 through 2.1.3.4 acceptable.

2.1.3.1 Table 2.2-1: Reactor Trip System Instrumentation Setpoints (pp. 2-4, 2-5, 2-7, 2-8 and 2-10)

A. Functional Unit 7, Overtemperature ΔT. Error terms Z, S, and the associated note revised to reflect new RTD instrumentation uncertainties, temperature streaming and the Westinghouse setpoint methodology. Note 2, Page 2-8, allowable value revised to 2.5% of ΔT span.

B. Note 1, Page 2-7, the reference to manifold instrumentation is deleted to agree with new RTD measurement system.

C. Functional Unit 8, Overpower ΔT. Error terms TA, Z, S revised to reflect new RTD instrumentation uncertainties, temperature streaming and the Westinghouse setpoint methodology.

D. Note 3, Page 2-10, the value for K6 has been increased. The K6 constant in the Overpower ΔT equation provides compensation for Tavg greater than nominal Tavg by reducing the overpower ΔT setpoint. The increase in uncertainties associated with RTD bypass removal increased the Technical Specification TA value. As a result, the licensee increased the safety analysis limit for K4 to allow the TS value for nominal K4 to remain unaffected. To account for this the margin in the Overpower ΔT setpoint equation for Tavg less than nominal Tavg was reduced and the value of K6 was increased to maintain the 118% thermal overpower limit.

In addition, the Note 2 allowable value was changed to 2.0% of span as a result of the RTD bypass removal uncertainties.

E. Functional Unit 12, Page 2-5, "Reactor Coolant Flow Low," the terms for Z and allowable value are modified to incorporate the modified RTD instrumentation instrument uncertainties.

2.1.3.2 Table 4.3-1: Reactor Trip System Instrumentation Surveillance Requirements (pp. 3/4 3-9, 3/4 3-13)

Functional Unit 7, "Overtemperature ΔT," the requirement to check RTD bypass loop flow has been deleted to be consistent with the replacement of the RTD bypass manifold system.

2.1.3.3 Bases 3/4 2.5: DNB Parameters (pp. B 3/4 2-4)

The licensee (supplement 1) increased the measurement error for RCS total flow rate from 2.1% to 2.4%. The increase in flow measurement uncertainty reflects the values documented in WCAP-13181 for RTD bypass removal. The 2.4% flow uncertainty also includes a 1% flow penalty to account for possible feedwater venturi fouling.

2.1.3.4 Specification 3/4.2.5: DNB Parameters (pp. 3/4 2-10)

Revised the surveillance requirements for the precision heat balance from prior to operation above 75% of rated thermal power after each refueling to prior to exceeding 95% of rated thermal power. Additionally, the DNB related parameter for reactor coolant system flow is increased from the current value of 391,000 gpm to a new value of 392,000 gpm by supplement 1 to the licensee submittal. The revised value of RCS flow reflects increased uncertainties for RTD bypass removal and 1% flow penalty for possible feedwater venturi fouling.

2.2 Reactor Systems Issues

Sections 2.2.1 through 2.2.3 discuss the review of reactor systems issues.

2.2.1 Current Method

The current method of measuring the hot and cold leg reactor coolant temperatures uses an RTD bypass system. The hot and cold leg temperature readings from each coolant loop are used for protection and control system inputs. The RTD bypass system was designed to address temperature streaming (non-uniform stratified flow in the cross-section) in the hot legs and, by use of shutoff valves, to allow replacement of the direct immersion narrow-range RTDs without draindown of the reactor coolant system (RCS). For increased accuracy in measuring the hot leg temperatures, sampling coops were placed in each hot leg at three locations of a cross-section, 120° apart. Each scoop has five orifices which sample that hot leg flow along the leading edge of the scoop. The flow from the scoops is piped to a manifold where a direct immersion RTD measures the average hot leg temperature of the flow stream from the three scoops in the hot leg. This bypass flow is routed back at a point downstream of the steam generator. The cold leg temperature is measured in a similar manner except that scoops are not used. This is because temperature streaming is not a problem due to the mixing action of the RCS pump.

2.2.2 New Method

The new method proposed for measuring the hot and cold leg temperatures includes the use of narrow range dual element RTDs, manufactured by WEED Company, which are mounted in thermowells to facilitate replacement without draindown of the RCS. The average hot leg temperature and the cold leg temperature are used to generate the reactor coolant loop differential temperature (ΔT) and average temperature (T_{avg}).

The hot leg temperature is measured using three of the WEED RTDs. Both elements of each hot leg RTD are wired to the appropriate process instrument rack where the second RTD input is a spare. The thermowells are located within two of the three existing RTD bypass manifold scoops, minimizing the need for additional hot leg piping penetrations. The third RTD will be located in an independent penetration nozzle. On loops A, B, and D the independent penetration nozzle is located in the same cross-sectional plane as the existing scoops, but offset 30° from the unused location. On loop C, the penetration nozzle will be relocated to a position approximately 12 inches upstream of the existing scoops at approximately 105° from top dead center. The unused scoops will be capped. The Weed RTDs are mounted to line up with the center hole of the five holes in the scoop. In the cases where new penetration nozzles are made the WEED RTDs will be inserted to the same depth as those in the scoops, which is the center hole depth.

Although unlikely, the RTD, or its electronics channel, can fail gradually, causing a gradual change in the loop temperature measurements. The licensee has committed to take regular temperature measurements to monitor RTD performance, so that any abnormal temperature shifts will be indicated.

An RTD failure will most likely result in an off scale high or low indication and will be detected through the existing control board T_{avg} and ΔT deviation alarms. If a failure of the RTD is diagnosed, two methods are available for addressing the failed RTD. Plant personnel can disconnect the failed element from the rack terminal strip and connect the other RTD element. Another option is for plant personnel to defeat the failed hot leg RTD and rescale the electronics to average the remaining two signals and incorporate a bias based upon the hot streaming measured in the loop.

One RTD will be located in each cold leg at the discharge of the reactor coolant pump. Again the existing RTD bypass penetration nozzle will be modified to accept the RTD thermowell. One element of the RTD will be considered active and the other element will be reserved as a spare. If a failure of a cold leg RTD is diagnosed, plant personnel can disconnect the failed element from the rack terminal strip and connect the other RTD element.

2.2.3 Analysis

The RTD response time is restricted with a technical specification Limiting Condition for Operation (LCO) to ensure consistency with assumptions in the accident analyses. The licensee presented information regarding the response time of the new RTD measurement system and also the accuracy of the new method for measuring the hot leg temperature which is discussed below.

2.2.3.1 RTD Response Time

The total response time for the current RTD bypass system and the proposed thermowell RTD system consist of the RTD bypass piping and thermal lag time, the RTD response time, and the electronic delay. The thermowell mounted RTDs have a response time equal to or better than the old bypass piping transport, thermal lag and direct immersion RTD. This allows the total RCS temperature measurement response time specified in technical specifications to remain unchanged at 6.0 seconds.

NUREG-0809 indicated that RTD response times have been known to degrade and that the Loop Current Step Response (LCSR) methodology is the recommended on-site method for checking RTD response times. The licensee has stated that they perform RTD response time testing, using the recommended LCSR method as stated in NUREG/CR-5560, for checking the RTD response time, which is acceptable to the staff.

Based on the above information the staff finds that the RTD response time has been addressed in an acceptable manner.

2.2.3.2 RTD Uncertainty

The following protection and control system parameters were affected by the change from one hot leg RTD to three hot leg RTDs; the Overtemperature delta T, Overpower delta T, Low RCS Flow rector trip functions, the RCS average temperature measurements used for control board indication and input to the rod control system, and the calculated value of the RCS flow uncertainty. System calculations were performed for each of the parameters and the results indicated that a sufficient margin exists to account for all known instrument uncertainties.

2.2.3.3 Non-LOCA Accidents

Only those transients which assume overtemperature delta-T (OTΔT) and overpower delta-T (OPΔT) protection function are potentially affected by changes in the RTD response time. As noted previously the new thermowell mounted RTDs have a response time equal to or better than that of the old bypass transport, thermal lag and direct immersion RTD. Because the total channel response time remains less than or equal to 6.0 seconds, it is concluded that the safety analysis assumption for the total OTΔT/OPΔT channel response time remains valid.

The change in RTDs has caused the uncertainty on T_{avg} to increase from +/-4°F to +/-5°F. However, +/-5°F is still less than the uncertainty previously assumed in the non-LOCA accident analysis.

The impact of the proposed increase in flow measurement uncertainty (i.e., 0.3%) on non-LOCA accident has also been considered. In this regard, the staff finds that the change in flow uncertainty has no impact because Thermal Design Flow is used in the non-LOCA accident analysis and the uncertainty is applied to the measured value of RCS flow.

The licensee determined that the RTD bypass elimination does not increase any uncertainty that will affect any initial condition assumed in any non-LOCA transient or the low primary coolant flow reactor trip function. Since the effect of the temperature response time is unaffected and the accuracy of the new system is bounded, the conclusions in Chapter 15 of the Seabrook FSAR remain valid.

2.2.3.4 LOCA Evaluation

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. However, the magnitude of the uncertainties are such that RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will be affected only slightly.

The T_{avg} uncertainty for Seabrook Unit 1 is now stated to increase to +/-5°F from +/-4°F. Therefore small peak cladding temperature (PCT) penalties have been applied to both the Large and Small Break LOCA analyses of record to address the T_{avg} uncertainty range increase. The PCT increases are 4°F for large break LOCA and 8°F for small break LOCA (Letter from T. C. Feigenbaum, North Atlantic Energy Service Corporation, to the NRC, "10 CFR 50.46 Annual Report," July 1, 1992). With these penalties the current PCT values become 2052.2°F for LBLOCA and 1981.2°F for SBLOCA. The PCT remains well below the regulatory limit of 2200°F. This is acceptable to the staff.

2.2.3.5 Precision Heat Balance

The licensee has also proposed to change TS surveillance requirement 4.2.5.3 to modify the performance of a precision heat balance which is used to determine RCS flow rate and to normalize the RCS flow instrumentation. Currently the TS requirements specify that the precision heat balance must be performed prior to operation above 75% of thermal rated power following each fuel loading. This calculation is performed each cycle to detect changes in the RCS flow element (elbow taps) characteristics that would affect the accuracy of the RCS flow indication.

The licensee proposes that the precision heat balance be performed prior to exceeding 95% thermal rated power to minimize the measurement uncertainties that are exacerbated at lower power levels. Performing the flow rate measurement prior to exceeding 95% rated thermal power provides a reasonable amount of excess margin to DNB in the highly improbable event that a degradation in RCS flow rate, masked by a simultaneous non-conservative change in all elbow taps, is not detected prior to reaching 95%. On this basis, the staff accepts the licensee's proposed change.

2.2.3.6 DNB Parameters

The licensee also proposed a change to TS 3.2.5 regarding DNB parameters. Currently the RCS flow rate is specified at greater than or equal to 391,000 gpm, which includes 2.1% flow uncertainty. The proposed change to 392,000 includes the thermal design flow of 382,800 gpm plus the cold leg elbow tap flow uncertainty of 2.4% flow. The 2.4% flow uncertainty includes 0.1% penalty for undetected feedwater venturi fouling. The staff finds the change in the flow rate acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Hampshire and Massachusetts State officials were notified of the proposed issuance of the amendment. The State officials had no comments.

4.0 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 (57 FR 30256). 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

To support the modifications required to eliminate the RTD bypass manifold system, the licensee proposed changes to the Seabrook Station TS. The TS revisions are a result of differences in the instrument system uncertainties between the thermowell mounted RTD system and the bypass manifold temperature measurement system. Evaluations performed by the licensee indicate that the instrument uncertainty values are acceptable. The impact of eliminating the RTD bypass system for Seabrook Station on FSAR Chapter 15 accidents has also been evaluated by the licensee. The review by the staff supports these conclusions. Since the RTD temperature response time and accuracy of the new system is not degraded, the former conclusions in the FSAR remain valid, and acceptable as described in Section 2.0.

The staff concludes that the modified RTD system is not functionally different from the current system except for the use of three RTDs instead of one in each hot leg. Based on the above, the staff finds that the proposed plant modifications to replace the RTD bypass manifold system with thermowell mounted, fast response, narrow range RTDs located directly in the reactor coolant system piping and the proposed TS changes are acceptable.

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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Date: August 10, 1992

AMENDMENT NO. 12 TO NPF-86 SEABROOK STATION DATED August 10, 1992

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R. Lobel
S. Brewer, 8 E23
C. Doutt, 8 H3