SEABROOK - UNIT 1

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	~~3.31~~ 3.5	~~[illegible]~~** 1.7 +~~0.47~~** 0.5	See Note 1	See Note 2
8. Overpower ΔT	~~4.8~~ 4.9	~~1.43~~ 2.2	~~0.12~~ 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 ~~1.04~~ 1.7 and the sensor error for Pressurizer Pressure is ~~0.47~~ 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.

TSC 92-01
Page 5

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	$\boxed{1.87}$ 1.9	0.6	≥90% of loop design flow*	$\boxed{\geq 89.4\%}$ ≥89.5% 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

TSC 92-01
PAGE 6

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 ~~Manifold~~ 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;

TSC 92-01
Page 7

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.

2.5%

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.0%~~ of ΔT span.

3@

TSC 92-01
PAGE 8

91

38

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

0.001386/°F

K_6 = ~~0.00128/°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 ~~3.4%~~ of ΔT span.

2.0%

342 91

40

TSC 92-01
PAGE 7

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature ~~detectors~~ (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the ~~loop~~ temperature ~~detectors~~, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

measurement system and temperature instrumentation delays

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the ~~loop~~ temperature ~~detectors~~, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

measurement system

TSC 92-01
Page 10

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}, ≤ 594.3°F

b. Pressurizer Pressure, ≥ 2205 psig*

c. Reactor Coolant System Flow, ≥ ~~391,000~~ 382,800 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 ~~75%~~ 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 2.1% flow measurement uncertainty.~~

4D4 91

SEABROOK - UNIT 1

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.	..	R(13)	N.A.	1, 2, 3*, 4*,
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, 5
7. Overtemperature ΔT	S	R~~(12)~~	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

SB3 91

101

TSC 92-01
PAGE 12

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

~~(12) Verify the RTD bypass loops flow rate.~~

(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.

SA6 91 105

3/4.2.5 DNB PARAMETERS

analytical limits

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.

pressure and 382,800 gpm for total RCS flow

uncertainty

The measurement ~~error of 2.1%~~ 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 flow rate measurement is performed prior to operation above 95% RTP to provide margin for flow degradation that is masked by changes in elbow tap normalization.

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.

III. Retype of Proposed Changes

See attached retype of proposed changes to Technical Specifications. The attached retype reflects the currently issued version of Technical Specifications. Pending Technical Specification Changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

Revision bars are provided in the right hand margin to indicate a revision to the text. No revision bars are utilized when the page is changed solely to accommodate the shifting of text due to additions or deletions.

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. Sourge Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6 x 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 THEMAL 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)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \frac{(1)}{(1 + \tau_6 S)} - T'] + K_3(P - P') - f_1(\Delta I)\}$$

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_0 = 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 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: (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.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature measurement system and temperature instrumentation delays (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the temperature measurement system and temperature instrumentation delays, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the temperature measurement system, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

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}, ≤ 594.3°F

b. Pressurizer Pressure, ≥ 2205 psig*

c. Reactor Coolant System Flow, ≥ 382,800 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.

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*, 4*
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, 5
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 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.

3/4.2.5 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 analytical limits 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 pressure and 382,800 gpm for total RCS flow are not exceeded.

The measurement uncertainty 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 flow rate measurement is performed prior to operation above 95% RTP to provide margin for flow degradation that is masked by changes in elbow tap normalization.

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.

IV. Safety Evaluation of License Amendment Request 92-01 Proposed Changes

New Hampshire Yankee is planning to implement a design change (DCR 90-03) at Seabrook Station during the second refueling outage. This design change will remove the existing Resistance Temperature Detector (RTD) Bypass System and replace this hot leg and cold leg temperature measurement method with a modified system consisting of fast-response thermowell mounted RTDs installed in the reactor coolant loop piping. The existing RTD Bypass System and the modified hot leg and cold leg temperature measurement system are described below. Westinghouse has prepared a topical report WCAP-13181 "RTD Bypass Elimination Licensing Report for Seabrook Nuclear Power Station" (Proprietary) in support of the four loop operation of Seabrook Station utilizing the new thermowell mounted RTDs. A copy of this report is provided in Section VIII. Yankee Atomic Electric Company (YAEC) has also evaluated the RTD Bypass System Elimination relative to containment response, Steam Generator Tube Rupture and Boron Dilution events. The Westinghouse and YAEC evaluation conclusions and documentation are discussed below and in Section V.

Existing RTD Bypass System

Currently, the hot leg and cold leg RTDs used for reactor control and reactor protection are inserted into manifolds in the Reactor Coolant System bypass loops. Separate bypass loops are provided for each reactor coolant loop such that individual loop temperature signals may be developed for use in the reactor control and reactor protection systems. A bypass loop from the hot leg side of each steam generator to the intermediate leg is used for the hot leg RTDs. Another bypass loop from the cold leg side of the reactor coolant pump to the intermediate leg is used for the cold leg RTDs. Both hot leg and cold leg manifolds empty through a common header to the intermediate leg between the steam generator and reactor coolant pump. The RTDs are located within manifolds and are inserted directly into the reactor coolant bypass flow without thermowells. The bypass manifold system limits high velocity coolant flow to the RTDs and compensates for the temperature streaming effects present in the hot leg piping. For each hot leg bypass loop, flow is provided by three scoop tubes located at 120 degree intervals around the hot leg. Because of the mixing effects of the reactor coolant pump, only one connection is required for bypass flow to the cold leg bypass manifold.

The output from the bypass loop RTDs provides the signal necessary to calculate the average loop temperature (T_{avg}) and the loop differential temperature (Delta T). The T_{avg} and Delta T signals are then input to the reactor protection system and the reactor control system.

Modified Hot Leg and Cold Leg Temperature Measurement System

The individual loop temperature signals required for input to the reactor control and reactor protection systems will be obtained using RTDs installed in each reactor coolant loop.

The hot leg temperature measurement on each loop will be accomplished using three fast response, narrow range, dual element RTDs mounted in thermowells. Both elements of each hot leg RTD are wired to the appropriate process protection rack where the second RTD input is a spare. To accomplish the sampling function of the RTD bypass manifold system and to minimize the need for additional hot leg piping penetrations, the thermowells will be located within two of the three existing hot leg RTD bypass manifold scoops. Due to a structural interference, the third RTD will be located in an independent boss. On loops A, B, and D the independent boss is located in the same cross-sectional plane as the existing scoops, but offset 30° from the unused location. On loop C, the boss will be relocated to a position approximately 12 inches upstream of the existing scoops at approximately 105° from top dead center. The unused scoops (the 120° location on loops A & C and the 240° location in loops B & D) will be capped. These 3 RTDs will be used to obtain the hot leg temperature used for generation of reactor coolant loop Delta T and T_{avg}.

This modification will not affect the single wide range RTD currently installed near the entrance of each steam generator. This RTD will continue to provide the hot leg temperature used for monitoring and control.

The cold leg temperature measurement on each loop will be accomplished using one fast response, narrow range, dual-element RTD located in each cold leg at the discharge of the reactor coolant pump (as replacements for the cold leg RTDs located in the bypass manifold). This RTD will measure the cold leg temperature which is used to calculate reactor coolant loop Delta T and T_{avg}. The existing cold leg RTD bypass penetration nozzle will be modified to accept the RTD thermowell. Both elements of the cold leg RTDs will be wired to the appropriate process protection rack where the second RTD input is a spare.

This modification will not affect the single wide range RTD in each cold leg currently installed at the discharge of the reactor coolant pump. This RTD will continue to provide the cold leg temperature for monitoring and control.

The RTD bypass manifold return line to the RCS crossover leg will be capped at the connection to the crossover leg.

WCAP-13181, Figure 1.3-1 provides a block diagram of the modified electronics. The hot leg RTD measurements (three per loop) will be electronically averaged in the reactor protection system. The hot leg averaging will be accomplished by additions to the existing process protection equipment. The averaged T_{hot} signal will then be used with the T_{cold} signal to calculate reactor coolant loop Delta T and T_{avg} which are used in the reactor control and reactor protection system.

The process protection equipment modifications will be qualified to the same level as the existing process protection equipment. The RTDs are environmentally qualified per New Hampshire Yankee's compliance with 10CFR50.49.

Existing control board Delta T and T_{avg} indicators and alarms provide the means of identifying RTD failures. Should the failure of a hot leg RTD be diagnosed, two methods are available for addressing the failed RTD. The preferred method is to utilize the second element of the RTD. Since both elements of each dual element RTD are wired to the appropriate process protection rack, I&C personnel can disconnect the failed element from the rack terminal strip and connect the other RTD element. If the spare element is not available, the second method is for the I&C 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 leg streaming measured in the loop. WCAP-13181, Appendix B provides the calculational methodology for hot leg temperature bias values. Should a failure of a cold leg RTD be diagnosed, the I&C personnel would disconnect the failed element from the rack terminal strip and connect the other RTD element.

The effect of the increased instrument uncertainty on updated Final Safety Analysis Report (UFSAR) Chapter 6 and 15 LOCA and non-LOCA accident analyses within the Westinghouse scope has been evaluated as discussed in WCAP-13181. Relative to both the LOCA and non-LOCA safety analyses, Westinghouse has concluded in WCAP-13181 that the modification does not affect the conclusions of the UFSAR safety analyses.

Additionally, Yankee Atomic Electric Company (YAEC) has evaluated the affect of the modified system for hot leg and cold leg temperature measurement on (1) containment response, (2) Boron Dilution events and (3) Steam Generator Tube Rupture design basis events.

Relative to containment response, YAEC concluded that during the limiting event (large break LOCA), the early containment pressure response during the blowdown phase may increase slightly due to the increase uncertain associated with the modification. However, the long term and peak containment pressure are still valid and the effects of the modification on the containment response is bounded by the current analysis. The YAEC evaluation of the affect of the modification on containment response is enclosed in Section VIII.

Yankee Atomic Electric Company has concluded that the increased uncertainty associated with the modification will have a negligible effect on the Steam Generator Tube Rupture analysis which was performed by them and submitted to the NRC on April 16, 1991 in NHY letter NYN-91061. Yankee Atomic Electric Company also concluded that the modification will have negligible effect on the Boron Dilution analysis to be performed by them for Cycle 3. The YAEC evaluation of the affect of the modification on the Steam Generator Tube Rupture analysis and on the Boron Dilution analysis which is to be performed for Cycle 3 is enclosed in Section VIII.

V. <u>Determination of Significant Hazards for License Amendment Request 92-01 Proposed Changes</u>

New Hampshire Yankee has determined that License Amendment Request 92-01 does not involve a significant hazard consideration pursuant to the standards of 10CFR50.92 based on the following evaluation.

1. The proposed changes do not involve a significant increase in the probability or consequences of a accident previously evaluated.

   Westinghouse has prepared WCAP-13181 "RTD Bypass Elimination Licensing Report for Seabrook Nuclear Station" (Proprietary) in support of the four loop operation of Seabrook Station utilizing new thermowell mounted RTD's. For the Westinghouse scope, WCAP-13181 contains a safety evaluation for this modified hot leg and cold leg temperature measurement system. This significant hazards evaluation addresses both the mechanical modifications to the reactor coolant system pressure boundary and the instrumentation uncertainty changes associated with the modified system.

   The installation of thermowells and fast response RTDs will not increase the probability of an accident previously analyzed. The modifications to the Reactor Coolant System pressure boundary will be performed utilizing the same ASME Section III installation requirements as were used for the original installation. The installation requirements are specified in the ASME Section III 1977 Edition thru Winter 1977 Addenda.

   The removal of the bypass piping and valves associated with this piping will enhance the integrity of the Reactor Coolant System. By removing significant lengths of piping, numerous valves and instrument penetrations the probability of a small break LOCA will be reduced.

   The new thermowell mounted RTDs have a total response time equivalent to the existing system as discussed in WCAP-13181. The increased instrumentation uncertainty associated with the new thermowell mounted RTDs necessitated an increase in the Overpower ΔT K4 term safety analysis limit and conservative changes to the K6 term to assure protection for all power ranges. The Overpower ΔT and Overtemperature ΔT functions thus continue to provide an equivalent degree of reactor protection. RTD signal processing and the added circuitry to the reactor protection system racks will be accomplished using the same type of Westinghouse 7300 series reactor protection system technology as has been previously qualified and used in the reactor protection system of Seabrook Station. There is no change in the use of the temperature signals by any reactor protection or reactor control system.

   The compliance of Seabrook Station to IEEE 279-1971, ("IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating

Stations"), applicable NRC General Design Criteria and regulatory guides has not changed.

This modification does not increase the radiological consequences of any accident previously evaluated. Although the pressure boundary will be modified, proper welding techniques, penetrant testing, radiographs, and system hydrostatic tests will insure the integrity of the pressure boundary and thus not contribute to any radiological consequences.

The proposed revisions to Technical Specification 3/4.2.5 (DNB parameters) for RCS flow from a value that includes measurement uncertainty to the analysis limit has no effect on the accident analyses since the analysis limit which is based on the thermal design flow will not be changed. Appropriate measurement uncertainties for the method used to measure RCS flow, including the effect of venturi fouling, have been determined. This uncertainty will be added to the RCS flow requirement of Technical Specification 3/4.2.5 to establish the acceptance criteria for the measured value of RCS flow. The acceptance criteria for the measured value of RCS flow will be specified in appropriate procedures.

Surveillance Requirement 4.2.5.3 for the precision heat balance determination of RCS flow is changed from being required prior to operation above 75% Rated Thermal Power (RTP) to being required prior to exceeding 95% RTP. Performance of the precision heat balance above 90% RTP was recommended by Westinghouse in association with the RTD bypass elimination to minimize flow rate measurement uncertainties that are exacerbated at lower power levels. The precision heat balance 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. Significant changes in the characteristics of all of the elbow taps over a single operational cycle is not credible. Performing the flow rate measurement prior to exceeding 95% RTP provides adequate margin to DNB in the highly improbable event that there is a degradation in RCS flow rate that is masked by a simultaneous non-conservative change in all elbow taps.

The effect of the increased instrument uncertainty on updated Final Safety Analysis Report (UFSAR) Chapter 6 and 15 LOCA and non-LOCA accident analyses within the Westinghouse scope has been evaluated as discussed in WCAP-13181. Relative to both the LOCA and non-LOCA safety analyses, Westinghouse has concluded in WCAP-13181 that the modification does not affect the conclusions of the UFSAR safety analyses.

Additionally, Yankee Atomic Electric Company (YAEC) has evaluated the affect of the modified system for hot leg and cold leg temperature measurement on (1) containment response, (2) Boron Dilution events and (3) Steam Generator Tube Rupture design basis events.

Relative to containment response YAEC concluded that during the limiting event (large break LOCA), the early containment pressure response during the blowdown phase may increase slightly due to the increased uncertainties associated with the modification. However, the long term and peak containment pressures are still valid and the effects of the modification on the containment response is bounded by the current analysis. The YAEC evaluation of the affect of the modification on containment response is enclosed in Section VIII.

Yankee Atomic Electric Company has concluded that the increase uncertainties associated with the modification will have a negligible effect on the Steam Generator Tube Rupture analysis which was performed by them and submitted to the NRC on April 16, 1991 in NHY letter NYN-91061. Yankee Atomic Electric Company also concluded that the modification will have negligible effect on the Boron Dilution analysis to be performed by them for Cycle 3. The YAEC evaluation of the affect of the modification on the Steam Generator Tube Rupture analysis and on the Boron Dilution analysis which is to be performed for Cycle 3 is enclosed in Section VIII.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The removal of the RTD Bypass System will not create the possibility of a new or different kind of accident from any accident previously evaluated. The reactor coolant pressure boundary modifications design and installation will be equivalent to the original RCS design and installation. Reactor coolant loop temperature inputs for reactor control and reactor protection functions will continue to be supplied. Other equipment important to safety will be unaffected and will continue to function as designed.

The removal of the Resistance Temperature Detector (RTD) bypass piping and the installation of a modified temperature measurement system does not affect the integrity of the reactor coolant system pressure boundary. This is due to the reactor coolant piping (pressure boundary component) modifications adhering to the ASME Code (Sections III, Class 1 and Section XI) and to the NRC General Design Criteria. Installation requirements will be equivalent to the original RCS installation pursuant to ASME Section III, 1977 Edition thru Winter 1977 Addenda.

The removal of the RTD Bypass System eliminates components that have been a major cause of plant outages in the industry as well as a major contributor to occupational radiation exposure. Additionally, with these components removed, the probability of a malfunction from them is eliminated. The installation of fast response thermowell mounted RTDs on the reactor coolant loop piping and additional processing electronics will continue to provide the individual loop temperature signals for input to the reactor control and reactor protection systems using components that are environmentally and seismically qualified.

The RTD Bypass System flow alarm is no longer required to warn of flow reduction that could affect instrument system response. Flow through the scoop tubes with thermowells is not monitored because blockage of the flow path is not credible. Blockage is not credible because of the multiple scoop tube holes, the size of the holes, and administrative and chemistry controls that prevent the introduction of objects that could block the flow path.

The modification does not affect the ability of the protection system to mitigate the radiological consequences of any accident. The new RTD signals are processed to provide equivalent signals to those provided by the original direct immersion RTDs. Since three RTDs will be used to provide an average hot leg temperature as opposed to the original use of one RTD, the consequences from a failed RTD are unchanged. Manual actions to bypass a failed RTD channel remain the same.

3. The proposed changes do not result in a significant reduction in the margin of safety.

The instrumentation uncertainty analysis associated with this modification has resulted in proposed Technical Specification changes to the uncertainty terms associated with Overpower ΔT and Overtemperature ΔT and low Reactor Coolant System (RCS) Flow reactor trip functions. Additionally RCS average temperature measurements used for control board indication and input to the rod control system, and the value of the RCS flow measurement uncertainty are also affected by the modification. The safety evaluations of this modification which have been performed by Westinghouse and YAEC referenced above conclude that sufficient margin exists such that margins to safety are not affected.

The proposed Technical Specification changes also include the elimination of the bypass piping loop low flow alarms and the revision to the Technical Specification requirement for RCS flow. The proposed change to the RCS flow requirement to specify analysis values provides consistency in this Technical Specification for DNB limits which currently specifies analysis values for Tavg and pressurizer pressure. This change to an analysis value for RCS flowrate does not affect any margin of safety.

The RTD Bypass System flow alarm is no longer required to warn of flow reduction that would affect instrument system response. Flow through the scoop tubes with thermowells is not monitored because blockage of the flow path is not credible. Blockage is not credible because of the multiple scoop tube holes, the size of the holes, and administrative and chemistry controls that prevent the introduction of objects that could block the flow path. The removal of this alarm does not result in a reduction in the margin of safety.

VI. Proposed Schedule for License Amendment Issuance and Effectiveness

New Hampshire Yankee requests NRC review of License Amendment Request 92-01 and issuance of a license amendment by July 1, 1992 (see enclosed License Amendment Request 92-01, Section VI). This schedule is proposed in support of NHY's plans to implement the RTD Bypass System Elimination design change during the second refueling outage which is scheduled to begin in September 1992.

VII. Environmental Impact Assessment

New Hampshire Yankee (NHY) has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, NHY concludes that the proposed change meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

VIII. Other Supporting Information

- Westinghouse Authorization Letter CAW-92-255 and accompanying affidavit

- Proprietary Information Notice

- Copyright Notice

- Westinghouse WCAP-13181 (Proprietary), "RTD Bypass Elimination Licensing Report for Seabrook Nuclear Station"

- Westinghouse WCAP-13193 (Non-Proprietary), "RTD Bypass Elimination Licensing Report for Seabrook Nuclear Station"

- Evaluation of the Effects of Removal of the RTD Bypass System on Containment Response of Seabrook Station

- Evaluation of the Effects of Removal of the RTD Bypass System on YAEC-1698 and the Seabrook Station Boron Dilution Analysis

---

Note: (1) New Hampshire Yankee notes that WCAP-13181 does not reflect NHY's proposed inclusion of the thermal design flow analysis value in Technical Specification 3.2.5c. WCAP-13181 specifies the RCS flow rate of 392,000 gpm which is associated with a revised RCS flow calorimetric uncertainty of 2.3% attributable to the new RTD hot leg and cold leg measurement system.

(2) New Hampshire Yankee notes that the deletion of the Technical Specification for low RCS Tavg coincident with reactor trip feedwater isolation Functional Unit (Table 3.3-4, Functional Unit 6.b) recommended in WCAP-13181 will be addressed in a future license amendment request.