January 20, 1988

Docket No. 50-423

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Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 60651, 66023 AND 66024)

The Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit No. 3, in response to your application dated September 9, 1987, and supplemental letters dated September 9, 1987, and September 30, 1987.

The amendment revises several Technical Specifications sections to accommodate the Cycle 2 reload, the installation of resistance temperature detectors in the reactor coolant system, and the restart of an isolated reactor coolant loop.

Enclosure 1 is the Safety Evaluation for the Millstone Unit 3 Cycle 2 reload and corresponding Technical Specification changes. The proposal includes analyses to justify a positive moderator coefficient below 70 percent power, ramping down to zero at 100 percent power. It is concluded that the reload analyses and the Technical Specification changes are acceptable. Future reload core cycles of Millstone 3 should provide an ATWS temperature coefficient (MTC) at equilibrium xenon conditions pending the staff evaluation of a forthcoming Westinghouse Owners Group response to a staff request for information on ATWS MTCs (letter A. Thadani (NRC) to R. Newton (Chairman, WOG), dated June 12, 1987.

Enclosure 2 is the Safety Evaluation for the Millstone Unit 3 proposed modifications to remove the resistance temperature detector (RTD) bypass manifold system and to replace RTD's located directly in the reactor coolant system hot and cold leg piping and corresponding Technical Specification changes. It is concluded that these modifications and Technical Specification changes are acceptable.

Enclosure 2 contains the Safety Evaluation for the new flow measurement uncertainty analysis which was also presented for review. This resolves the open item in TAC 60651. You have cleaned the feedwater venturi meters used for the RCS flow measurement calibration prior to the forthcoming Cycle 2 operation and have committed (Ref. 10) to modify the Technical Specification to indicate that a 0.1% penalty will be applied to the flow measurement uncertainty if the venturi meters are not cleaned upon each refueling. The modified Technical Specification should be submitted prior to Cycle 3 operation.

The notice of issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

original signed by

Robert L. Ferguson, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
Amendment No. 12 to NPF-49.
Safety Evaluation - Cycle 2
Safety Evaluation - RTD

cc w/enclosures:
See next page

LA:PDI-4	PM:PDI-4	D:PDI-4	OGC
SNorris	RFerguson:bd	JStolz	
01/15/88	01/15/88	01/15/88	01/ /88

Mr. E. J. Mroczka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 3

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

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.*

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. NPF-49

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated September 9, 1987, and supplemental letters dated September 9, 1987, and September 30, 1987 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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*Northeast Nuclear Energy Company is authorized to act as agent and representative for the following Owners: Central Maine Power Company, Central Vermont Public Service Corporation, Chicopee Municipal Lighting Plant, City of Burlington, Vermont, Connecticut Municipal Electric Light Company, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, The Village of Lyndonville Electric Department, Western Massachusetts Electric 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-49 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, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 20, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILTIY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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

Remove	Insert
2-5	2-5
2-6	2-6
2-8	2-8
2-9	2-9
2-11	2-11
B2-5	B2-5
3/4 1-4	3/4 1-4
3/4 1-11	3/4 1-11
3/4 1-12	3/4 1-12
3/4 2-15	3/4 2-15
3/4 2-18	3/4 2-18
3/4 2-24	3/4 2-24
3/4 3-8	3/4 3-8
3/4 3-10	3/4 3-10
--	3/4 3-10a
3/4 3-14	3/4 3-14
3/4 3-28	3/4 3-28
3/4 3-30	3/4 3-30
3/4 3-34	3/4 3-34
3/4 4-8	3/4 4-8
3/4 5-1	3/4 5-1
3/4 5-9	3/4 5-9
3/4 6-14	3/4 6-14
3/4 9-1	3/4 9-1
--	3/4 9-1a
B3/4 1-3	B3/4 1-3
--	B3/4 1-3a
B3/4 2-5	B3/4 2-5
B3/4 2-6	B3/4 2-6
B3/4 2-7	B3/4 2-7
B3/4 9-1	B3/4 9-1
5-5	5-5

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					
1) Four Loops Operating	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
2) Three Loops Operating	7.5	4.56	0	≤ 80% of RTP**	≤ 82.1% 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 contant ≥ 2 seconds	≤ 6.3% of RTP** 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**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^{+5} cps	≤ 1.4 x 10^{+5} cps
7. Overtemperature Δ T					
a. Four Loops Operating	8.3	5.76	1.67 + 1.17 (Temp + Press)	See Note 1	See Note 2
b. Three Loops Operating	12.0	5.77	1.73 + 1.17 (Temp + Press)	See Note 1	See Note 2
8. Overpower Δ T	4.8	1.22	1.67	See Note 3	See Note 4

*Loop design flow = 94,600 gpm (Four Loops Operating); 99,600 (Three Loops Operating)
**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low	5.0	1.77	3.3	≥ 1900 psia	≥ 1890 psia
10. Pressurizer Pressure-High	5.0	1.77	3.3	≤ 2385 psia	≤ 2395 psia
11. Pressurizer Water Level-High	8.0	5.13	2.7	≤ 89% of instrument span	≤ 90.7% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.52	0.78	≥ 90% of loop design flow*	≥ 89.1 % of loop design flow*
13. Steam Generator Water Level Low-Low	20.5	18.98	1.75	≥ 23.5% of narrow range instrument span	≥ 22.6% of narrow range instrument span
14. General Warning Alarm	N.A.	N.A.	N.A.	N.A.	N.A.
15. Low Shaft Speed - Reactor Coolant Pumps	3.8	0.5	0	≥ 97.8% of rated speed	≥ 94.6% of rated speed
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.

**RTP = RATED THERMAL POWER

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	$\leq$ 12.1% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8					
1) Four Loops Operating	N.A.	N.A.	N.A.	$\leq$ 37.5% of RTP**	$\leq$ 39.6% of RTP**
2) Three Loops Operating	N.A.	N.A.	N.A.	$\leq$ 37.5% of RTP**	$\leq$ 39.6% of RTP**
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	$\leq$ 51% of RTP**	$\leq$ 53.1% of RTP**
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	$\geq$ 10% of RTP**	$\geq$ 7.9% of RTP**
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.
21. Three Loop Operation Bypass Circuitry	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

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

Where:

ΔT	=	Measured ΔT by Reactor Coolant System 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, $\tau_1 = 12$ s, $\tau_2 = 3$ 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	=	1.08 (Four Loops Operating); 1.01 (Three Loops Operating);
K_2	=	0.01313;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	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}, $\tau_4 = 33$ s, $\tau_5 = 4$ 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 (Continued)

NOTE 1: (Continued)

T'	$\leq$	587.1°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000663/psi;
P	=	Pressurizer pressure, psia;
P'	=	2250 psia (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 such that:

(1) For $q_t - q_b$ between -30% and + 10%, $f_1(\Delta I) = 0$, 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 -30%, the ΔT Trip Setpoint shall be automatically reduced by 3.6% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% 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 2.1% ΔT span (Four Loop Operation); 3.6% ΔT span (Three Loop Operation).

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

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}, τ_7 = 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 = 0.00129/°F for T > T" and K_6 = 0 for T ≤ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 587.1°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 2.8% ΔT span.

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. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Trip System.

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, 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.

Operation with a reactor coolant loop out of service requires Reactor Trip System modification. Three loop operation is permissible after resetting the K1 input to the Overtemperature ΔT channels, reducing the Power Range Neutron Flux High setpoint to a value just above the three loop maximum permissible power level, and resetting the P-8 setpoint to its three loop value. These modifications have been chosen so that, in three loop operation, each component of the Reactor Trip System performs its normal four loop function, prevents operation outside the safety limit curves, and prevents the DNBR from going below 1.30 during normal operational and anticipated transients.

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."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer Water Level High trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Flow Low trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 38% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.6% Δk/k, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6300 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than $+0.5 \times 10^{-4}$ $\Delta k/k/^{o}F$ for the all rods withdrawn, beginning of cycle life (BOL) condition for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 $\Delta k/k/^{o}F$ at 100% RATED THERMAL POWER.

b. Less negative than -4.0×10^{-4} $\Delta k/k/^{o}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only**.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the above limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

---

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 6700 gallons,

2) A boron concentration between 6300 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 250,000 gallons,

2) A minimum boron concentration of 2300 ppm, and

3) A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 23,620 gallons,

2) A boron concentration between 6300 and 7175 ppm, and

3) A minimum solution temperature of 67°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 1,166,000 gallons,

2) A boron concentration between 2300 and 2600 ppm,

3) A minimum solution temperature of 40°F, and

4) A maximum solution temperature of 50°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1.6% Δ k/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 385,210 gpm, and

b. $F^N_{\Delta H} \leq 1.49 \left[1.0 + 0.3 (1.0 - P)\right]$

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable in-core detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement, and

3) The measured value of RCS total flow rate shall be used since uncertainties of 1.8% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that $F^N_{\Delta H}$ and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.1.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.1.2, is assumed to exist.

4.2.3.1.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.1.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.1.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

POWER DISTRIBUTION LIMITS

RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

THREE LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.2 The indicated Reactor Coolant System (RCS) total flow rate and F^N_H shall be maintained as follows:

a. RCS total flow rate $\geq$ 304,780 gpm, and

b. $F^N_{\Delta H} \leq 1.351 \left[1.0 + 0.43 (1.0 - P)\right]$

Where:

1) $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.2b. takes into consideration a measurement uncertainty of 4% for incore measurement, and

3) The measured value of RCS total flow rate shall be used since uncertainties of 2.0% for flow measurement have been included in Specification 3.2.3.2a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 32% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 37% of RATED THERMAL POWER within the next 4 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

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 Each of the parameters of Table 3.2-1 shall be verified to be within its limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS	
	Four Loops in Operation	Three Loops in Operation & Loop Stop Valves Closed
Indicated Reactor Coolant System T_{avg}	$\leq$ 591.2°F	$\leq$ 583.4°F
Indicated Pressurizer Pressure	$\geq$ 2226 psia*	$\geq$ 2226 psia*

*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 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. When the Minimum Channels OPERABLE requirement is met, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of the Turbine Control Valves.

ACTION 13 - With one of the diverse trip features (undervoltage or shunt trip attachments) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N/A
2. Power Range, Neutron Flux	$\leq$ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A
6. Source Range, Neutron Flux	N.A.
7. Overtemperature Δ T	$\leq$ 7 seconds*
8. Overpower Δ T	$\leq$ 7 seconds*
9. Pressurizer Pressure--Low	$\leq$ 2 seconds
10. Pressurizer Presssure--High	$\leq$ 2 seconds
11. Pressurizer Water Level--High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Reactor Coolant Flow--Low	
a. Single Loop (Above P-8)	≤ 1 second
b. Two Loops (Above P-7 and below P-8)	≤ 1 second
13. Steam Generator Water Level--Low-Low	≤ 2 seconds
14. Low Shaft Speed-Reactor Coolant Pumps	≤ 0.6 second**
15. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
16. Safety Injection Input from ESF	N.A.
17. Reactor Trip System Interlocks	N.A.
18. Reactor Trip Breakers	N.A.
19. Automatic Trip and Interlock Logic	N.A.
20. Three Loop Operation Bypass Circuitry	N.A.

**Speed sensors are exempt from response time testing. Response time of the speed signal portion of the channel shall be measured from detector output or first electronic component in the channel.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACUTATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Nuetron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q(17)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Nuetron Flux, High Positive Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux High Negative Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
5. Intermediate Range,	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, 17)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature Δ T	S	R	Q(17)	N.A	N.A	1, 2
8. Overpower Δ T	S	R	Q(17)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(17, 18)	N.A.	N.A.	1

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACUTATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10. Pressurizer Pressure--High	S	R	Q(17, 18)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q(17)	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R	Q(17)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) (Not used)

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of less than or equal to 5 times background.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) (NOT USED)

(13) Reactor Coolant Pump Shaft Speed Sensor may be excluded from CHANNEL CALIBRATION.

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

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

(16) Automatic undervoltage trip.

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

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-3	3.3	1.01	1.75	≤ 8.0 psig	≤ 8.8 psig
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	3.3	1.01	1.75	≤ 3.0 psig	≤ 3.8 psig
d. Steam Line Pressure--Low	17.7	15.31	2.2	≥ 658.6 psig*	≥ 644.9 psig*
e. Steam Line Pressure - Negative Rate--High	5.0	0.5	0	≤ 100 psi/s**	≤ 122.7 psi/s**

TABLE 3.3-4 (Continued)

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIPS SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A	N.A.	N.A.	N.A.	N.A
b. Steam Generator Water Level--High-High (P-14)	3.7	2.33	1.75	≤ 82.0% of narrow range instrument span.	≤ 82.8% of narrow range instrument span.
c. Safety Injection Actuation Logic	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Valves.				
d. T_{ave} Low Coincident with Reactor Trip (P-4)					
1) Four Loops Operating	N.A.	N.A.	N.A.	≥ 564°F	≥ 560.6°F
2) Three Loops Operating	N.A.	N.A.	N.A.	≥ 564°F	≥ 560.6°F
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low					
1) Start Motor-Driven Pumps	20.5	18.98	1.75	≥ 23.5% of narrow range instrument span.	≥ 22.6% of narrow range instrument span.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pumps	20.5	18.98	1.75	≥ 23.5% of narrow range instrument span.	≥ 22.6% of narrow range instrument span.
d. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps	N.A.	N.A.	N.A.	≥ 2800V	≥ 2720V
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA Trip Setpoints and Allowable Values.				
7. Control Building Isolation					
a. Manual Actuation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	N.A.	N.A.
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.3	1.01	1.75	≤ 3.0 psig	≤ 3.8 psig
e. Control Building Inlet Ventilation Radiation	N.A.	N.A.	N.A.	≤ 1.5×10^{-5} μc/cc	≤ 1.5×10^{-5} μc/cc
f. Outside Chlorine High	N.A.	N.A.	N.A.	≤ 5 ppm	≤ 5 ppm

TABLE 3.3-4 (Continued)

ENGINEERING SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIPS SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	N.A.	≥ 2800 volts with a ≤ 2 second time delay.	≥ 2720 volts with a ≤ 2 second time delay.
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	N.A.	N.A.	≥ 3710 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation.	≥ 3706 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation.
9. Engineering Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1985 psig	≤ 1995 psig.
b. Low-Low T_{avg}, P-12	N.A.	N.A	N.A.	≥ 553°F	≥ 549.6°F
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level, P-14	See Item 5 above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(5)}/37^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generators	≤ 12
b. Steam Line Isolation	$\leq 6.8^{(3)}$
5. Containment Pressure--High-3	
a. Quench Spray	$\leq 32^{(2)}/42^{(1)}$
b. Phase "B" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
c. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
d. Service Water	$\leq 90^{(1)}$
6. Containment Pressure--High-2	
a. Steam Line Isolation	$\leq 6.8^{(3)}$
7. Steam Line Pressure - Negative Rate--High	
a. Steam Line Isolation	$\leq 6.8^{(3)}$
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	$\leq 6.8^{(3)}$
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. Loss-of-Offsite Power	
a. Motor-Driven Auxiliary Feedwater Pump	≤ 60

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
11. Loss of Power	
a. 4 kV Bus Undervoltage (Loss of Voltage)	$\leq$ 13
b. 4 kV Emergency Bus Undervoltage (Grid Degraded Voltage)	$\leq$ 18(7)/310(8)
12. T_{ave} Low Coincident With Reactor Trip (P-4)	
a. Feedwater Isolation	$\leq$ 12(3)
13. Control Building Inlet Ventilation Radiation	
a. Control Building Isolation	$\leq$ 3.7
14. Outside Chlorine High	
b. Control Building Isolation	$\leq$ 7

ISOLATED LOOP

LIMITING CONDITION FOR OPERATION

3.4.1.5 The RCS loop stop valves of an isolated loop shall be shut and the power removed from the valve operators.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied: either shut the loop stop valves and remove power from the valve operators within one hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.5 The RCS loop stop valves of an isolated loop shall be verified shut and power removed from the valve operators at least once per 31 days.

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops,

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops, or greater than 2300 ppm whichever is less

c. The isolated portion of the loop has been drained and is refilled, and

d. The reactor is subcritical by at least 1.6%Δk/k.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The reactor shall be determined to be subcritical by at least 1.6% Δk/k within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.3 Within 4 hours prior to opening the loop stop valves, the isolated loop shall be determined to:

a. Be drained and refilled, and

b. Have a boron concentration greater than or equal to the boron concentration of the operating loops, or greater than 2300 ppm whichever is less

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6618 and 6847 gallons,

c. A boron concentration of between 2200 and 2600 ppm, and

d. A nitrogen cover-pressure of between 636 and 694 psia.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks to be within the above limits, and

2) Verifying that each accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution; and

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2300 and 2600 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 50°F.

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

ACTION

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

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

ACTION:

With one Recirculation Spray System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Recirculation Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 130 psid when tested pursuant to Specification 4.0.5;

c. At least once per 18 months by verifying that on a CDA test signal, each recirculation spray pump starts automatically after a 660 ± 20 second delay;

d. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal; and

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Spray Additive System shall be OPERABLE with:

a. A chemical addition tank containing a volume of between 18000 and 19000 gallons of between 2.41 and 3.10% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical addition tank to each Containment Quench Spray subsystem pump suction.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis is within the above limits.

c. At least once per 18 months, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2300 ppm.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6300 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2300 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.1.3 Valve 3CHS-V305 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

REFUELING OPERATIONS

BORON CONCENTRATION

Limiting Condition for Operation

3.9.1.2 The boron concentration of the Spent Fuel Pool shall be maintained uniform and sufficient to ensure that the boron concentration is greater than or equal to 800 ppm.

Applicability

During ALL fuel assembly movements within the spent fuel pool.

Action

With the boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool.

Surveillance Requirements

4.9.1.2 Verify that the boron concentration is greater than or equal to 800 ppm prior to any movement of a fuel assembly into or within the spent fuel pool, and every 72 hours thereafter during fuel movement.

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.6% Δ k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 21,020 gallons of 6300 ppm borated water from the boric acid storage tanks or 1,166,000 gallons of 2300 ppm borated water from the refueling water storage tank (RWST). A minimum RWST volume of 1,166,000 gallons is specified to be consistent with ECCS requirement.

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.6% Δ k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4100 gallons of 6300 ppm borated water from the boric acid storage tanks or 250,000 gallons of 2300 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The minimum RWST solution temperature for MODES 5 and 6 is based on analysis assumptions in addition to freeze protection considerations. The minimum/maximum RWST solution temperatures for MODES 1, 2, 3 and 4 are based on analysis assumptions.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control

HEAT FLUX HOT CHANNEL FACTOR AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, in maintained within the limits.

$F^N_{\Delta H}$ 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.

The $F^N_{\Delta H}$ as calculated in Specifications 3.2.3.1 and 3.2.3.2 are used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.
The difference between the three and four-loop $F^N_{\Delta H}$ equations is due to a more restrictive $F^N_{\Delta H}$ used in the safety analyses for three-loop operation. In four-loop operation, the allowable measured $F^N_{\Delta H}$ calculated in Specification 3.2.3.1 at 65% Rated Thermal Power is $\leq$ 1.65. In three-loop operation, however, $F^N_{\Delta H}$ is restricted to a measured value $\leq$ 1.55 to be consistent with the safety analyses for three loop operation. At zero power, both specifications allow the same measured $F^N_{\Delta H}$.

Fuel rod bowing reduces the value of DNB ratio. 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.

HEAT FLUX HOT CHANNEL FACTOR AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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 ($F_{xy}^{RTP^Q}$) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors of 1.8% for four loop flow and 2.0% for three loop flow for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detectd could bias the result fromthe precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi will be added if venturis are not verified clean every 18 months. Any fouling which might bias the RCS flow rate measurement greater than 0.5% 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 indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of defined in Specification 3.2.3.1 and 3.2.3.2.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

QUADRANT POWER TILT RATIO (Continued)

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4 2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are 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. The indicated T_{avg} values are 591.2°F (four loop operations) or 583.4°F (three loops operating) and the indicated pressurizer pressure value is 2226 psia (four loop or three loop operation). The calculated values of the DNB related parameters will be an average of the indicated values for the operable channels.

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. Measurement uncertainties have been accounted for in determining the parameter limits.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2300 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.1.2 Boron Concentration in Spent Fuel Pool

The limitations of this specification ensure that in the event of a fuel assembly handling accident involving either a misplaced or dropped fuel assembly, the K_{eff} of the spent fuel storage racks will remain less than or equal to .95.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) refueling machines will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum nominal enrichment of 3.4 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal enrichment of 3.8 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 61 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 95.3% hafnium and 4.5% natural zirconium. All control rods shall be clad with stainless steel.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,240 cubic feet at a nominal T_{avg} of 587°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-3.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% $\Delta k/k$ for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal 10.35-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 756 PWR fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 12

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated September 9, 1987 (Ref. 1), the Northeast Nuclear Energy Company (NNECo) (the licensee) proposed to amend Appendix A of Facility Operating License No. NPF-49. The requested amendment furnished information to support authorization for Millstone Unit 3 operation during Cycle 2.

The Millstone 3 Cycle 2 (hereafter referred to as M3C2) reload will consist of 84 fresh assemblies in place of 65 Region 1 and 19 Region 2 assemblies from the previous cycle. In support of the M3C2 reload NNECo submitted a topical report (Ref. 2) which summarizes the reload scope, the plant transient analyses, and the design and safety analyses. In addition, Technical Specification (TS) changes were proposed to allow operation with a positive moderator coefficient at power levels less than 70 percent, ramping down to zero at 100 percent power. The licensing reports for the latter TS changes are provided in Reference 3. Additional TS changes for Cycle 2 startup were submitted in Reference 4.

In a related matter, NNECo submitted a Radial Peaking Factor Limit Report for Cycle 2 of operation in Reference 6. Also, in response to a staff request for additional information (Ref. 7), the licensee provided additional information (Ref. 8) regarding the expected moderator temperature coefficient for Millstone 3 during the upcoming Cycle 2.

2.0 EVALUATION

2.1 Cycle 2 Reload Description

The M3C2 reload will retain 109 Westinghouse (W) assemblies from the initial cycle and will add 84 W assemblies into an established scatter pattern. The fresh assemblies have minor mechanical differences in pellet, sleeve, and end plug characteristics which do not impact the design bases and do not require TS changes. The new fuel has a higher enrichment than the initial fuel. The Cycle 2 core loading inventory is given in Table 2.1 of this SE. The reload

evaluation is based on the Cycle 1 exposure of 17500 to 19000 MWD/MTU and a Cycle 2 exposure of 16000 MWD/MTU. The reload evaluation was performed using accepted methodology described in WCAP-9273 (Ref. 5). Necessary changes to the TS for the Cycle 2 reload are limited to those related to the increased enrichment (TS Section 5.3.1) and the proposed positive moderator coefficient as discussed in the following SE Item 2.2.

Submittal of the Radial Peaking Factor Report (Ref. 6) is required by Technical Specification 6.9.1.6 and identifies nuclear aspects of the design calculations for Millstone 3 Cycle 2. The report identifies F_{xy} limits to provide assurance that the initial conditions assumed in the LOCA analysis are met. The determination of the limits is performed in accordance with acceptable procedures described in WCAP-8403 "Power Distribution Control and Load Following Procedures" which is part of the approved reload methodology for Westinghouse core reloads. We find the identified limits acceptable,

Table 2.1

Millstone 3 Cycle 2 Core Loading Inventory

Region Designation	Number of Assemblies	Initial Enrichment (w/o U235)	BOC Burnup Ave (MWD/MTU)	EOC Burnup Ave (MWD/MTU)
2	45	2.899	19600	35600
3	64	3.395	14200	30200
4A	56	3.5	0	16000
4B	28	3.8	0	16000

2.2 Moderator Temperature Coefficient

The proposed amendment includes a request to change the Millstone 3 moderator temperature coefficient Technical Specification. The request proposes to increase the upper bound of the moderator coefficient given in Specification 3.1.1.4 from zero pcm/deg F to +5 pcm/deg F for power levels below 70 percent of rated power (One pcm is equal to a reactivity change of 10^{-5} delta-k/k) . Above 70 percent power the allowed value would decrease linearly to zero pcm/deg F at full power. Approval of the change is requested for both N and N-1 loop operation. The licensee's justification is provided in a Licensing Report (Ref. 3) prepared by Westinghouse Electric Corporation.

The licensee has assessed the impact of a positive moderator coefficient on the accident analyses presented in Chapter 15 of the Millstone Unit 3 FSAR. Those incidents which were found to be sensitive to positive or near-zero moderator coefficients were reanalyzed. These incidents are limited to transients which cause the reactor coolant temperature to increase. Accidents not reanalyzed include those resulting in excessive heat removal from the reactor coolant system and those for which a large negative moderator coefficient is more limiting. We agree with the licensee's conclusions about which transients did and did not require reanalysis.

The transients not reanalyzed are:

A. RCCA misalignment

B. Startup of an inactive reactor coolant pump

C. Excessive heat removal due to feedwater system malfunctions

D. Excessive load increase event

E. Steam generator tube rupture

F. Main steam line depressurization

G. Feedwater system pipe break

H. Inadvertent operation of the ECCS at power

The incidents reanalyzed, with two exceptions, used a +5 pcm/deg F moderator temperature coefficient, assumed to remain constant for variations in temperature. The reanalyses were done with the same computer codes used in the Millstone 3 FSAR analyses and with the same or more conservative uncertainty allowances. The incident exceptions are the rod ejection and rod withdrawal from a subcritical condition, for which the computer code cannot accept a constant coefficient.

The incidents reanalyzed and their results are:

A. Uncontrolled RCCA bank withdrawal from a subcritical condition

The results of the reanalysis of this transient produced a peak heat flux which did not exceed the full power nominal value presented in the FSAR. Therefore the conclusions presented in the FSAR are still applicable.

B. Uncontrolled RCCA bank withdrawal at power

The results of the reanalysis of this transient show that the nuclear flux and overtemperature delta-T trips prevent the core minimum DNBR ratio from falling below 1.3 for this incident, so that the conclusions presented in the FSAR are still valid.

C. Loss of Reactor Coolant Flow

The most severe loss of flow transient is caused by the simultaneous loss of power to all four reactor coolant pumps. For this case a minimum DNBR was obtained which is above the limit of 1.3.

D. RCP Shaft Seizure

The locked rotor event was reanalyzed because of the potential effect of the positive MTC on the nuclear power transient and thus on the peak RCS pressure and fuel temperature. A positive MTC will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. For three loops initially operating the peak clad temperature was 1828 deg F and the peak RCS pressure was 2613 psia; these values do not exceed the accepted safety limits.

E. Loss of External Electrical Load/Turbine Trip

The turbine trip event was reanalyzed for the beginning of cycle (BOC) and at end of cycle (EOC). Four cases for both N-loop and N-1 loop operation were analyzed: reactor at beginning-of-life with operation of the pressurizer spray and pressurizer power operated relief valves (PORV) and the reactor at end-of-life with no credit for pressurizer spray or PORVs. The result of a turbine trip is a core power which momentarily exceeds the secondary system power removal causing an increase in RCS coolant temperature. The reactivity addition due to a positive MTC causes an increase in both nuclear power and RCS pressure. Since the DNB ratio does not drop below its initial value and the peak RCS pressure increases slightly but is less than 110 percent of design the staff's acceptance criteria is met and the conclusions presented in the FSAR are still applicable.

F. RCCA Ejection

The results of reanalysis of this incident show the fuel and clad temperature do not exceed the limits specified in the FSAR, so that the conclusions presented in the FSAR are still valid.

G. Loss of Normal Feedwater

The results of reanalysis of this incident show minimal changes in the pressurizer water volume and reactor coolant system temperature so that the conclusions presented in the FSAR are still valid.

H. Inadvertant Opening of a Pressurizer Safety or Relief Valve

The results of reanalysis of this incident show that the minimum DNBR stays above the accepted safety limit and the conclusions presented in the FSAR are still valid.

I. Anticipated Transients Without Scram

Although the Millstone 3 Cycle 2 Technical Specification moderator temperature coefficient (MTC) is being increased to + 5 pcm/deg F below 70% power and to a linear variation from + 5 pcm/deg F at 70% power to zero pcm/deg F at 100% power, the licensee stated, in its response (Ref. 8) to a staff request for additional information (Ref. 7), that the Cycle 2 core design, hot full power (HFP) MTC will be more negative than - 7.6 pcm/deg F with equilibrium xenon conditions. In addition, the MTC for Cycle 2 will become more negative as cycle burnup increases. This Cycle 2 HFP MTC at equilibrium xenon conditions is more conservative than the - 5.5 pcm/deg F value used in generic ATWS studies (Ref. 9) performed for four-loop Westinghouse plant designs and which yielded a limiting peak pressure of 3200 psig. Since Millstone 3 is, from an ATWS point of view, similar to the four-loop class of Westinghouse plants for which the studies were performed, the staff concludes that ATWS considerations are not significantly impacted. In addition the Millstone 3 Cycle 2 physics startup tests will provide a hot, zero power (HZP) MTC reference point which can be used to assess the core design HFP MTC with equilibrium xenon conditions.

Results

Since the reanalysis of plant transients did not result in exceeding any of the limits specified in the existing analyses for the Millstone Unit No. 3 reactor, we conclude the proposed Technical Specification change will not result in any significant loss of safety margins, and is therefore acceptable.

2.3 Increase in Allowable Value of Boron Concentration in Borated Water Sources

The incorporation of a positive moderator coefficient and the anticipated longer fuel cycle necessitates increasing the required boric acid in the Emergency Core Cooling System (ECCS) accumulators and the refueling water storage tank (RWST). The licensee's submittal includes a safety evaluation by Westinghouse of the impact of raising the boron concentration on the Loss of Coolant Accident (LOCA) and non-LOCA analyses and design considerations (Ref. 3). The proposed amendment would increase the RWST boron concentration from 2000 ppm to the range 2300 to 2600 ppm and the ECCS accumulator boron concentration from 1900 ppm into the range 2200 to 2600 ppm. The changes are necessary to assure that the reactor will remain subcritical in cold shutdown following a LOCA when higher enrichment fuel is used for anticipated extended fuel cycles for Millstone Unit 3. This increase also has implications for the analysis of several non-LOCA events and for the system chemistry through the mechanism of pH change in the post accident containment sump liquid inventory.

Non-LOCA Safety Analyses

The only non-LOCA events which are affected by the increased boron concentration are those for which the ECCS is actuated. Each of these events has been examined to determine the effect of increased boron concentration. The results show that the increased boron has a generally helpful effect on the individual event and in no case does it have a harmful effect.

LOCA Analysis

The small break LOCA analysis makes no assumption about the boron concentration in the ECCS water (shutdown is achieved and maintained by control rods). Thus the increased boron concentration has no effect on the small break LOCA analysis.

During the initial portion of the large break LOCA analysis subcriticality is maintained by voids in the core and the increased boron concentration has no effect on this portion of this event. Since the peak clad temperature occurs during this portion of the event there is no effect on the results of the LOCA analysis. Based on the similarity of assumptions to the LOCA analyses, the increase in ECCS water boron concentration would also have no effect on the long term mass and energy releases to the containment.

During the initial portion of the large break LOCA analysis subcriticality is maintained by voids in the core and the increased boron concentration has no effect on this portion of this event. Since the peak clad temperature occurs during this portion of the event there is no effect on the results of the LOCA analysis. Based on the similarity of assumptions to the LOCA analyses, the increase in ECCS water boron concentration would also have no effect on the long term mass and energy releases to the containment.

For post LOCA shutdown no credit is taken for control rods in the Millstone 3 analysis. Thus the boron in the ECCS water is relied upon to maintain shutdown. The ECCS water, when mixed with other sources (reactor coolant water, etc.) must produce a boron concentration sufficient to maintain the reactor in a shutdown state. The adequacy of the increased concentration to achieve this goal is to be addressed for each reload.

Other Considerations

Increasing the ECCS boron concentration reduces the pH of the containment spray and recirculating core coolant solutions. The reduction in pH can lead to reduction of the iodine spray removal coefficient and decontamination factor (DF), an increase in the rate of hydrogen production due to zinc corrosion and an increase in the potential for chloride induced stress corrosion cracking of stainless steel. These effects have been examined by the licensee with the following results.

Since the FSAR analysis of radiological consequences of the large-break LOCA does not assume containment spray iodine removal, there is no change in the FSAR conclusion in this area.

Examination of zinc corrosion rate data shows that the corrosion rate for the minimum pH resulting from the increased boron concentration is less than that assumed in the FSAR. This is acceptable. Corrosion of other materials (e.g., aluminum) decreases monotonically with decreasing pH.

The pH range resulting from the boron increase would be outside the range recommended to minimize chloride stress corrosion cracking of stainless steel. To bring the post LOCA recirculation sump pH into an acceptable range, the sodium hydroxide concentration in the chemical addition tank (CAT) has been increased. The proposed TS changes (LCO 3.6.2.3) reflect this concentration increase along with a reduction in solution volume and are acceptable.

2.4 Additional Changes to Support the Cycle 2 Reload

In Reference 4 the licensee provided a safety evaluation for the following TS changes which are intended for incorporation in this amendment.

(1) Section 3/4 4.1.6 - The LCO and Surveillance Requirement for boron concentration in an isolated loop is revised to limit the required concentration to a maximum of 2300 ppm.

The present LCO requires that an isolated loop be at a boron concentration greater than or equal to the unisolated portion of the reactor coolant system (RCS). The proposed change would allow the isolated loop to be brought into service as long as the water used to fill the isolated loop has a boron concentration greater than the rest of the RCS. The licensee has evaluated the relevant design basis events for consequences of the change and has concluded that the existing FSAR analyses are still bounding or are not affected. The staff finds this change acceptable.

(2) Section 3.9.1.2 - The LCO and Surveillance Requirement for boron concentration during refueling operations were changed to require 800 ppm boron in the spent fuel pool subcriticality such that a dropped or misplaced fuel assembly will not cause the k-effective of the pool to exceed 0.95. The licensee has reviewed the relevant design basis events and has concluded that the increase in boron will not adversely affect the consequences. This change is therefore acceptable.

(3) Section 5.3.1 - The Design Features description of the fuel assemblies was revised to delete the maximum total weight limitation on individual fuel rods and clarifies the maximum enrichment for future core reloads. The change identifies the maximum nominal enrichment as 3.8 weight percent U-235. This is consistent with the proposed Cycle 2 design and is acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

The following Millstone Unit 3 Technical Specification changes have been proposed for operation during Cycle 2:

A. Changes to identify and support a positive moderator coefficient

TS Page	TS LCO or BASES	Title
3/4 1-4	3.1.1.3	Moderator Temperature Coefficient
3/4 1-11	3.1.2.5	Borated Water Source - Shutdown
3/4 1-12	3.1.2.6	Borated Water Sources - Operating
3/4 5-1	3.5.1	Accumulators
3/4 5-9	3.5.4	Refueling Water Storage Tank
3/4 6-14	3.6.2.3	Spray Additive System
3/4 9-1	3.9.1	Boron Concentration
B 3/4 1-3	3/4.1	Boration Systems
B 3/4 9-1	3/4.9.1	Boron Concentration

B. Other Technical Specification Changes

5-5	5.3.1	Fuel Assemblies
3/4 9-1c	3.9.1.2	Boron Concentration
3/4 4-8	3.4.1.6	Isolated Loop Startup

The above replacement TS pages identified in the licensee's request (Attachment 2 to letter B12661 and Attachment 1 to letter B12692) are acceptable as proposed.

4.0 SUMMARY

We have reviewed the reports submitted for the Cycle 2 operation of Millstone Unit 3. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. Sufficient basis has been provided to allow reload of 109 W assemblies and the use of a positive moderator coefficient. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

5.0 ENVIRONMENTAL CONSIDERATION

This 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 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 20, 1988

Principal Contributors:

M. McCoy
D. Fieno

7.0 REFERENCES

1. Letter, B-12661, E. J. Mroczka, NNECo, to U. S. NRC, dated September 9, 1987, with attachments

2. Reload Safety Evaluation for Millstone Unit 3 Cycle 2, Westinghouse Electric Corporation, August 1987

3. Positive Moderator Coefficient Licensing Report for Millstone Unit No. 3, Westinghouse Electric Corporation, August 1987

4. Letter, B-12692, E. J. Mroczka, NNECo, to U. S. NRC dated September 30, 1987, with attachments

5. WCAP-9293-A, "Westinghouse Reload Safety Methodology", dated July 1985

6. Letter, MP-11104, S. E. Scace, NNECo, to W. Russell, Region I Administrator, dated November 10, 1987

7. Letter, R. L. Ferguson (NRC) to E. J. Mroczka (NNECo), dated December 1, 1987.

8. Letter, E. J. Mroczka (NNECo) to R. L. Ferguson (NRC), dated January 7, 1988.

9. Letter (NS-EPR-83-2833 from E. P. Rahe W to S. J. Chilk (NRC), dated October 3, 1983.

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

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 12

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated September 9, 1986 (Ref. 1), the Northeast Nuclear Energy Company (NNECo) (the licensee) indicated that the reactor coolant temperature measurement system for the hot and cold legs for Millstone Unit 3 will be modified and requested changes to the plant's Technical Specifications (TS). This modification is to eliminate the Resistance Temperature Device (RTD) bypass manifold to improve availability, reduce radiation exposure, and reduce maintenance. However, the new hot leg temperature measurement method has the disadvantage of a slightly longer response time. Reference 1 included the proposed TS changes and also WCAP-11496 (Proprietary), the licensing report (Ref. 2) with RCS nozzle and thermowell locations, results of the reanalysis of affected FSAR Chapter 15 accidents, and a flow measurement uncertainty analyses.

2.0 BACKGROUND

The current method of measuring the hot and cold leg reactor coolant temperatures uses an RTD bypass system. This system was designed to address temperature streaming 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 scoops were placed in each hot leg at three locations of a cross section, 120 degrees apart. The flow from the scoops is piped to a manifold where a direct immersion RTD measures the average temperature of the flow from the three scoops. This bypass flow is routed back to the RCS downstream of the steam generator. The cold leg temperature is measured in a similar manner with piping to a bypass manifold except that no scoops are used, as temperature streaming is not a problem due to the mixing action of the RCS pump.

The new method proposed for measuring the hot and cold leg temperatures uses narrow-range fast response RTDs manufactured by Weed Instruments, Inc. The RTDs are placed in thermowells to allow replacement without draindown. The thermowells, however, increase the response time.

The RTDs for the hot legs in loops B and C are placed within the existing scoops for Millstone Unit 3. A hole will be drilled through the end of each scoop so that water will flow through the existing holes in the leading edge of the scoop, past the RTD, and out through the hole. The RTD measures the temperature at one point in the new method. This is in contrast to the temperature measurement of the average of the flow from the five sample holes from the hot leg scoops used in the RTD bypass flow method. However, in the new method, the radial location of each RTD measurement is at the same radius as the center hole of the scoop. Therefore, the licensee states, it is the equivalent of the average scoop sample if a linear radial temperature gradient exists in the pipe.

The RTDs in the hot legs for loops A and D have two of the three RTDs mounted in thermowells as described above. The third thermowell cannot be installed in the existing scoop location due to structural interference. Therefore, this thermowell is located downstream of the existing scoop in independent bosses. Although these RTDs are not in scoops, the sensor will be at the same radial location (in line with the center hole) as the other RTDs which are mounted inside the existing scoops.

The design for the measurement of the cold leg temperature has also been modified. A single thermowell with one fast response, narrow range, dual-element RTD is mounted in each cold leg. One of the dual elements is a spare. This is in place of the original method in which the measurement was by an external RTD in the cold leg bypass manifold.

An electronic system is used to perform the averaging of the reactor coolant hot leg signals from the three RTDs in each hot leg and then to transmit the signal for the average hot leg temperature to protection and control systems. There is a routine for performing a quality check of the three temperature signals for each hot leg, which are used to get an average value of the temperature variation. Also, there is a capability to add a bias to the averaging calculation, if needed, to compensate for the loss of one of the three RTD sensor inputs. The bias considers the past history of the previous hot leg readings.

Reference 1 provided the TS changes regarding the new reactor coolant system (RCS) temperature measurement system modifications required because of the elimination of the RTD bypass loop. Reference 2 provided the results of the reanalysis of several Chapter 15 non-LOCA accidents. The accuracy of the hot leg temperature was included in an RCS flow measurement uncertainty analysis submitted in Reference 2. Since Millstone Unit 3 has N-1 loop capability, information was presented for both four and three loop operation where applicable.

3.0 ANALYSIS

We questioned the licensee regarding the response time and uncertainty effects of the new measurement system. The licensee responded to our questions in a letter dated November 25, 1987 (Ref. 3). The increased response time has the primary impact on the results of the accident analysis. The uncertainty of the hot leg temperature measurement affects the accident analysis and is the principal contributor in the analysis for calculating the RCS flow measurement uncertainty.

3.1 RTD Response Time

The overall response time of the new thermowell RTD temperature system is one second longer than the former RTD bypass system (7.0 vs 6.0 seconds). The 7.0 second overall response time for the new RTD system is a result of adding a 1.5 second electronics delay to the 5.5 second response time for the RTD sensor. The single RTDs in loops A and D not inside of scoops are expected to have a slightly faster response time. However, no credit is taken for this. Because of the increased channel response time, there are longer delays from the time when fluid conditions in the reactor coolant system (RCS) require overtemperature delta-T or overpower delta-T reactor trips until a trip is actually generated. The licensee presented information in Reference 2 concerning the FSAR Chapter 15 non-LOCA accidents that rely on the above mentioned trips and which were evaluated for the longer response time.

As noted in NUREG-0809, (Ref. 4), extensive RTD testing has revealed RTD time response degradation with aging. In view of this, surveillance tests are needed. The approved in situ method for measuring RTD response time is the Loop Current Step Response (LCSR) method. The licensee cannot tell at present if there is margin from the manufacturer's stated response time for the RTD sensor of 5.5 seconds. This will be determined after the initial installed response time test and succeeding tests which will show any effect of possible drift in response time as mentioned in Reference 4.

3.2 RTD Uncertainty

The new method of measuring each hot leg temperature with three thermowell RTDs manufactured by Weed Instruments, Inc., used in place of the RTD bypass system with three scoops, has been analyzed to be slightly more accurate. The new RTD thermowell with measurement at one point may have a small streaming error relative to the former scoop flow measurement because of a temperature gradient over the 5-inch scoop span. However, this gradient has been calculated to have a small effect. Also, since possible temperature uncertainties from imbalanced scoop flows are eliminated, the overall result is more effective. In addition, since the new method uses three RTDs for each hot leg temperature measurement, it is statistically a more accurate temperature measurement than the former method which used only one RTD for each hot leg temperature measurement. Although the new hot leg RTD temperature measurement is initially slightly more accurate than with the former RTD manifold method, it becomes less accurate because of the additional uncertainties introduced when the signals are processed for averaging before being sent to the 7300 processing system. System uncertainty calculations were performed by the licensee that verify that sufficient allowance has been made in the reactor protection system setpoints to account for an increased initial RCS average temperature error. Therefore, the current values of nominal setpoints for the Millstone Unit 3 TS are still valid.

The licensee has stated (Ref. 10) that during the Cycle 2 startup test program they will collect the new hot leg and cold leg narrow range RTD temperature measurements at 100% of power and compare the Tavg and delta T measurements to those values obtained during the Cycle 1 startup test program before elimination of the RTD bypass system. This comparison will be utilized to

verify that the new method of temperature measurement is accurate. Similar temperature measurements are also being taken at the Catawba plant to compare their values before and after a similar modification.

In response to questions regarding the monitoring of the RTDs for failure, the licensee responded (Ref. 3) that the following three types of alarms are in place:

		HI/LO Deviation Alarm
1.	Tavg/Auctioneered Tavg	± 2° F
2.	Tref/Auctioneered Tavg	± 3° F
3.	Delta T/Auctioneered Delta T Deviation	± 2° F

These alarms are monitored continuously and a channel operability check is performed every 12 hours. Information is also obtained on individual RTDs to obtain bias values that can be used to compensate for a failed RTD in the hot leg averaging calculation. This information is obtained on a monthly basis.

In response to a question on RTD drift, the licensee explained (Ref. 3) their method for calibrating the RTDs at each refueling prior to startup. This method uses a Westinghouse recommended in-situ calibration known as the "Incore Thermocouple and Resistance Temperature Detector (RTD) Cross Calibration". Data is collected during plant heat-up following refueling. At RCS temperatures of approximately 320° F, 390° F, 450° F, and 515° F, the resistance of each of the 32 RTDs is recorded, as well as the indicated temperature from each incore thermocouple. These readings are recorded 4 times at each temperature within a span for 10 minutes. The calibration of the digital voltmeter used to measure resistance is verified and the data is validated prior to recommencement of heat-up. Prior to reactor startup, the calibration of each RTD and each incore thermocouple is verified by comparison to the average temperature indicated by the RTDs. If any RTD deviates from its calibration curve, a new calibration curve is generated based upon the data obtained during the cross-calibration. In addition, each incore thermocouple is compared to a calculated average temperature and must be within 2° F to be considered operable. The average temperature of the RTDs is compared to the incore thermocouples which are considered to be very stable and to have no drift. If a systematic, long-term drift in RTD calibration occurred, it would mean that an abnormally large number of incore thermocouples would be found with a deviation greater than 2° F compared to the RTDs. Steps would then be taken to correct the RTDs for drift.

The platinum resistance temperature sensors (RTDs) are believed to be very stable and to have relatively small calibration drifts. However, according to several sources (Refs. 5, 6, 7) RTDs have been known to shift in calibration. Therefore at the time of taking the calorimetric heat balance at refueling, necessary steps for correction (recalibration in a lab) should be made for any appreciable calibration drifts encountered or the RTD(s) declared inoperable and replaced. For small deviations found in their in-situ cross calibration method, the licensee stated (Ref. 3) that the calibration of the resistance to voltage converters of the affected RTD(s) will be adjusted to account for the shift. The stated drift of the RTDs by the manufacturer is ± 0.2° F per year. This is significantly less than the value assumed for the uncertainty calculations for the protection system.

3.3 Non-LOCA Accidents Reanalyzed

The primary impact of the RTD bypass system elimination is the increased RTD response time. Thus, only those events which rely on the overtemperature and overpower delta-T (OTDT and OPDT) reactor trips are impacted. The accidents in FSAR Sections 15.1 to 15.6 were examined by the licensee (Ref. 1 and 2) and the following non-LOCA accidents affected by the longer RTD response time were reanalyzed: (1) the Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal; (2) Loss of Load/Turbine trip; (3) Inadvertent Opening of a Pressurizer Safety or Relief Valve; and (4) the Steamline Rupture at Power. These accidents are described in Chapter 15 of the FSAR.

The licensee stated that the LOFTRAN computer code was used for the analysis of these events. For each event reanalyzed the basic assumptions regarding initial conditions, instrument errors, and setpoint errors that are not directly related to RTD Bypass Elimination remain largely the same as those in Chapter 15 of the FSAR. However, certain additional changes were made. The ± 30 psi allowance on pressurizer pressure in FSAR Section 15.0.3.2 was increased to ± 45 psi for more conservatism. Also increased uncertainties were applied to pressurizer and steam generator water levels which were increased from 5% to 5.73% and 5.53% respectively. These increased uncertainties were incorporated to bound calculated increases in the associated transmitter uncertainties. Also, changes were made in the reactor protection system setpoints to account for the new thermowell mounted RTDs. The time constant for lead/lag compensation of delta-T was increased from 8.0 to 12.0 seconds for the non-LOCA transient analysis.

The first accident, Uncontrolled RCCA Withdrawal, is described in Section 15.4.2 of the FSAR. For this event, the High Neutron Flux and Overtemperature delta-T reactor trips are assumed to provide protection against DNB. This event was analyzed with the increased time constants and the lead/lag changes of delta-T. Plots of DNBR versus time were provided which showed that the DNBR criterion was met for this accident.

The Loss of Load/Turbine Trip event is described in Section 15.2.3 of the FSAR. This event is affected by the increase of RTD response time which provides input to the over temperature delta-T trip. Both N and N-1 cases were reanalyzed and for all combinations of reactivity feedback and pressure control. The DNBR limit of 1.3 was met and the primary system pressure remained below 110% of the design value of 2,500 psi.
The Inadvertent Opening of a Pressurizer Safety Relief Valve event is described in Section 15.6.1 of the FSAR. The reanalysis was done considering both N and N-1 loop operation. The positive moderator coefficient causes nuclear power to increase as pressure decreases until reactor trip occurs at Overtemperature delta T. The DNBR remains above 1.30 throughout the transient and is therefore acceptable.

For the Steamline Rupture at Power event the analysis included the increased response time and was performed for both N and N-1 loop operation. The analysis showed that the design basis as described in WCAP-9226-RI has been met.

In response to a question regarding the impact of the RTD response time on the uncontrolled boron dilution at power event the licensee responded (Ref. 3) that the event would not be affected. The licensee stated that boron dilution analyses at power are performed to show that sufficient operator action time is available to terminate the dilution prior to loss of the shutdown margin requirement specified in the TS (1.6% delta-k/k). The calculated operator action time is not affected by the replacement of the RTDs since it is defined as starting simultaneously when reactor trip occurs. Therefore, the available operator action time would not change. For the case with manual rod control and no operator action to terminate the dilution, the power and temperature rise will result in a trip on an over temperature delta-T signal. The boron dilution transient results in a positive reactivity insertion and is essentially equivalent to an uncontrolled rod withdrawal at power transient. The reactivity insertion rate is within the range analyzed in WCAP-11496. Those analysis results are applicable to the boron dilution transient at power and thus show that the acceptance criteria for the boron dilution event are met with the increased RTD response time.

In summary, the impact of the RTD bypass elimination for Millstone Unit 3 on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated. For the events impacted, it was demonstrated that the conclusions presented in the FSAR (including DNBR remains above 1.30) remain valid for both N and N-1 loop operation.

3.4 LOCA Evaluation

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The licensee stated in Reference 2 that 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 not be affected. Past sensitivity studies concluded that the inlet temperature effect on peak clad temperature is dependent on break size. As a result of these studies, the LOCA analyses are performed at a nominal value of inlet temperature without consideration of small uncertainties. The RCS flow rate and steam generator secondary side temperature and pressure are also determined using the loop average temperature (Tavg) output. These nominal values used as inputs to the analyses are not affected due to the RTD bypass elimination. It is concluded that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint for both N and N-1 loop operation without requiring any reanalysis.

3.5 Flow Measurement Uncertainty

The licensee provided a new flow measurement analysis in Reference 2 that accounts for changes due to the RTD bypass removal. The methodology used was the same as that used for the Shearon Harris Unit 1 plant as provided in WCAP-11169, Rev. 1, October 1986. This analysis used the plant-specific instrumentation for the Millstone Unit 3. The results of the analysis indicated that for four loop operation the flow measurement uncertainty can be

reduced from the current value of ± 2.4% (not including a 0.1% penalty for feedwater venturi fouling allowance) to a new value of ± 1.8% (including the cold leg elbow taps and excluding feedwater venturi fouling). For three loop operation the results of the analysis indicated that the flow measurement uncertainty can be reduced from the current value of ± 2.76% to ± 2.00%, excluding the 0.1% penalty for feedwater venturi fouling. Our review has found the results of the analysis to be acceptable.

The licensee provided information in TS section 3/4.2.3 which showed that the minimum indicated RCS flow rate is 385,210 gpm for four loop operation and 304,780 gpm for three loop operation. The corresponding flow measurement uncertainties are ± 1.8% and ± 2.0% for four and three loop operation respectively. The minimum indicated RCS flow rate values are obtained by increasing the thermal design flows (TDFs) of 378,400 gpm (4 x 94,600 gpm/loop) and 298,800 gpm (3 x 99,600 gpm/loop) for four and three loop operation respectively by their corresponding flow measurement uncertainties.

The licensee is installing inspection ports (Ref. 8) upstream of the venturis. This is to be similar in design to that provided in another plant (Ref. 9). The inspection port is about 13 inches upstream of the low pressure tap of the venturi meter. This places the inspection port about midway between the low pressure and high pressure taps. With a 4-3/4" diameter opening of the inspection port, the observer will have an unobstructed view of the inside pipe wall opposite the port and can see the inlet contour of the venturi along the opposite wall. With the aid of an inspection mirror and light, the entire circumference of the inside pipe can be viewed as well as the converging section of the venturi. To inspect the diverging section of the venturi a flexible fiber-scope would be used.

As stated above, for the new analysis there is a further reduction in flow measurement uncertainty for four loop operation of Millstone Unit 3 from ± 2.4% to ± 1.8%. It is the staff's position that the additional ± 0.1 penalty for venturi fouling should be applied unless the venturi meters are cleaned at each refueling. The minute buildups on the venturi that could affect the flow measurement cannot be accurately quantified by just visual means.

When the 0.1% venturi fouling factor is added, the resulting flow measurement uncertainties are ± 1.9 for four loop operation and ± 2.1% for three loop operation. The TS minimum indicated RCS flow rate for these conditions is 385,590 gpm for four loop operation and 305,075 gpm for three loop operation.

The licensee has stated (Ref. 3) that the latest Millstone Unit 3 RCS measured flow rate is 415,294 gpm based on three of the loops having a flow rate of 110% of thermal design flow and one loop at 109% of thermal design flow. This is equivalent to an indicated RCS flow rate (415,294 gpm) that is 109.75% of thermal design flow (1.0975 x 378,400 gpm). Therefore the existing Millstone Unit 3 RCS flow rate is well above the required thermal design flow rate.

TS sections 4.2.3.1.6, 4.2.3.2.6 and the bases for TS section 3/4.2.4 (page B 3/4 2-6) will need to be modified to state that the penalty for undetected fouling of the feedwater venturis of 0.1% will be added to the flow measurement uncertainty values if the venturis are not cleaned. This is to be done before the precision heat balance is made to calibrate the RCS flow rate indicators

(approximately once per 18 months). The licensee has stated that the feedwater venturis have been cleaned for the Cycle 2 operation. The licensee has stated (Ref. 10) that the above TS's will be modified to reflect the requirement of 0.1% penalty if the venturis are not cleaned and submitted for NRC approval. The staff requires this modification prior to Cycle 3 operation.

3.6 Instrumentation

A. Current System

Currently, the hot and cold leg RTD's are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTD's and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTD's. The RTD's are located in manifolds and are directly inserted into the reactor coolant bypass loop without thermowells. Each RTD manifold (one hot leg and one cold leg per reactor coolant loop) contains two narrow-range RTD's: one for protection and control system inputs and one as a spare. Flow into each bypass loop is provided by three scoops located at 120° intervals around the hot leg and a tap into the corresponding cold leg.

Each loop's pair of RTD's (one in the hot leg and one in the cold leg) is used to provide inputs for protection system functions based on the average loop temperatures ($Tavg = T_{HOT} + T_{COLD})/2$) and the loop differential temperature $\Delta T = T_{HOT} - T_{COLD}$). Protection functions based on these inputs are: overtemperature ΔT and overpower ΔT reactor trips with their associated (non-protection) rod stop and turbine runback actions, low Tavg main feedwater isolation, and low-low Tavg (P-12) steam dump block signals.

Each loop's pair of RTD's is also used to provide inputs for control system functions based on the average loop temperature and the loop differential temperature. Control functions based on these inputs are: turbine loading stop from auctioneered low Tavg; rod, steam dump and pressurizer level control from auctioneered high Tavg; rod insertion limit alarms from auctioneered high ΔT and Tavg.

B. Modified System

The hot leg temperature inputs from each reactor coolant loop will be developed from three fast response, narrow range single-element RTD's mounted in thermowells located within the three existing RTD bypass manifold scoops (except for loops A and D where the two of the three thermowells will be mounted in the scoops with the third thermowell located downstream in an independent boss). One fast response, narrow range dual-element RTD per loop will be mounted in a thermowell located at the existing penetration for the bypass loop into the cold leg. Both elements of the dual-element RTD will be wired to the process instrumentation cabinets with one element per RTD serving as a spare. In the event of an element failure, switchover to the spare element can be readily accomplished.

Each hot leg temperature input for protection system functions will be developed by electronically averaging the signals from the three new fast response, narrow range RTD's. This averaged input will replace the single input from the currently installed hot leg RTD. Each cold leg input for

protection system functions will be provided by the new fast response, narrow range RTD which replaces the currently installed cold leg RTD. In the event of a hot leg RTD failure, the electronics allow a bias developed from historical data for the failed RTD to be manually added via a potentiometer to the remaining two RTD signals in order to obtain an average value comparable to the three-RTD average prior to failure of the one RTD. If an element in the dual-element cold leg RTD fails, the spare element can be used instead.

Inputs for the control system functions will be provided, through isolators, from the average loop temperatures and loop differential temperatures calculated by the protection system. This aspect of the design has not been changed; only the use of three hot leg RTD's instead of one per loop to provide a hot leg temperature is different.

Our review and evaluation is based upon Sections 7.2 and 7.3 of the Standard Review Plan (SRP). Those sections state that the objectives of the review are to confirm that the reactor trip and engineered safety features actuation system satisfy the requirements of the acceptance criteria and guidelines applicable to the protection system and will perform their safety function during all plant conditions for which they are required. Since our review indicates that the modified system does not functionally change (except three hot leg RTD's are utilized instead of just one) the reactor trip and engineered safety features actuation systems, the staff's original evaluation conclusions for these systems, as documented in Section 7.2 and 7.3 of the SER for Millstone Nuclear Power Station Unit 3, (NUREG-1031), remain valid. Based on this and the licensee's statement that the new hardware for the RTD bypass elimination has been qualified to WCAP-8587, "Methodology to Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," we find the plant modifications to eliminate the RTD bypass manifold and to install fast response RTD's directly in the reactor coolant system hot and cold legs to be acceptable.

3.7 Mechanical Design

The removal of the RTD bypass loop and the installation of the RTD thermowells require modifications to the hot leg scoops, the hot leg piping, the crossover leg bypass return nozzle, and the cold leg bypass manifold connection. The licensee proposed that welding and non-destructive examinations will be per ASME Code Section XI requirements. Fabrication will be in accordance with the ASME Code Section III. By the letter of December 23, 1987, the licensee has verified that during installation there was no deviation from the proposed RTD configurations in the referenced report. We find the welding, non-destructive examinations and fabrication to be acceptable.

4.0 EVALUATION OF TECHNICAL SPECIFICATIONS

As a result of the modifications associated with the removal of the existing RTD bypass manifold and replacement by fast response RTDs, changes to the plant's TSs (Ref. 3) were proposed to allow operation of Millstone Unit 3 with the RTD bypass manifolds removed and to incorporate the reduced flow measurement uncertainty values for N and N-1 operation.

Change 1 - Bases TS Section 2.1, page B 2-5, for Overtemperature delta-T - The statement "transit delays from the core to the temperature detectors (about 4 seconds)." The about 4 seconds was deleted as the bypass piping is removed. This is acceptable as an editorial change to represent the present condition.

Change 2 - Bases TS Section 2.1, page B 2-6, for Overpower delta-T - The statement "compensation for piping delays from the core to the loop temperature detectors" was eliminated. This change is an editorial change to reflect the present condition.

Change 3 - TS Section 3.2.3.1, page 3/4 2-15 for four loop operation - The indicated RCS total flow rate was changed from 387,500 gpm to 385,210 gpm. The flow measurement uncertainty was changed from 2.4% to 1.8%. These changes are acceptable as discussed in the Analysis Section.

Change 4 - TS Section 3.2.3.2, page 3/4 2-18 for three loop operation - The indicated RCS total flow rate was changed from 307,050 gpm to 304,780 gpm. The flow measurement uncertainty was changed for 2.76% to 2.0%. These changes are acceptable as explained in the Analysis Section.

Change 5 - Table 3.2-1, page 3/4 2-24, DNB Parameters - The indicated RCS Tavg values for four loop and three loop operation were changed from 589.2° F and 581.7° F respectively to 591.2° F and 583.4° F respectively. The indicated pressurizer values for four loop and three loop operation were changed from 2,220 psia to 2,226 psia. These changes reflect the changes in uncertainty from replacing the RTD bypass system and are acceptable.

Change 6 - Table 3.3-2, page 3/4 3-8, Reactor Trip System Instrumentation Response Times - The response times for Function Unit 7, Overtemperatures delta T and Functional Unit 8, Overpower delta-T was changed from 4.0 to 7.0 seconds. These changes are due to the removal of the RDT bypass system and the longer response time of the new RTD thermowell system. They are acceptable as explained in the Analysis Section.

Change 7 - Table 4.3-1, page 3/4 3-10, Reactor Trip System Instrumentation Surveillance Requirements - Under "channel calibration" for Functional Unit 7, Overtemperature delta T, the reference to note 12 was deleted. Also note 12 was revised to read "(NOT USED)". There changes are acceptable as they are editorial changes to reflect the present condition.

As a result of the new instrumentation associated with the removal of the existing RTD bypass manifold and replacement by fast response RTD's, the following changes to the plant's TS were proposed:

Change 1 - Change the entries for Z and Sensor Error for Functional Unit 7.a, Overtemperature ΔT, Four Loops Operating, in Table 2.2-1 from "5.9" to "2.3" to "5.76" and "1.67 + 1.17 (Temp + Press)" respectively.

Change 2 - Change the entries for Z and Sensor Error for Functional Unit 7.b, Overtemperature ΔT, Three Loops Operating, in Table 2.2-1 from "5.9" and "2.3" to "5.77" and "1.73 + 1.17 (Temp + Press)" respectively.

CHANGE 3 - Change the entries for Z and Sensor Error for Functional Unit 8, Overpower ΔT, in Table 2.2-1 from "1.43" and "0.11" to "1.22" and "1.67" respectively.

CHANGE 4 - Change the entries for Z, Sensor Error and Allowable Value for Functional Unit 12, Reactor Coolant Flow-Low, in Table 2.2-1 from "1.74," ".8" and "89.3" to "1.52," "0.78" and "89.1" respectively.

CHANGE 5 - In Note 1 to Table 2.2-1, change "ΔT = Measured ΔT by RTD Manifold Instrumentation" to "ΔT = Measured ΔT by Reactor Coolant System Instrumentation."

CHANGE 6 - Change the entry for τ_1 in Note 1 to Table 2.2-1 from "8s" to "12s."

CHANGE 7 - Change the allowable value (for Overtemperature ΔT) in Note 2 to Table 2.2-1 from "4.1%" to "3.6%" for three loop operation.

CHANGE 8 - Change the allowable value (for Overpower ΔT) in Note 4 to Table 2.2-1 from "3.4%" to "2.8%."

CHANGE 9 - Change the response times for Functional Unit 7, Overtemperature ΔT and Functional Unit 8, Overpower ΔT, in Table 3.3-2 from "4.0" to "7.0."

CHANGE 10 - Under "CHANNEL CALIBRATION" for Functional Unit 7, Overtemperature ΔT, in Table 4.3-1, delete reference to Note 12. Also reivse Note 12 to read "(NOT USED)."

CHANGE 11 - Change the entries under Allowable Value for Functional Units 5.d.1 and 5.d.2 from "562°F" to "560.6°F."

CHANGE 12 - Change the entry under Allowable Value for Functional Unit 9.b, Low-Low Tavg, P-12, in Table 3.3-4 from "549.7°F" to "549.6°F."

CHANGE 13 - Change the entry for the response time for Functional Unit 12.a from "6.8" to "12.".

Changes 1, 2, 3, 4, 7, 8, 11, and 12 above are new values based on revised instrumentation uncertainties resulting from the bypass manifold elimination. The values were calculated using the Westinghouse setpoint methodology as previously approved by the staff for Millstone Unit 3 TS (see SER for Millstone 3, NUREG-1031, Section 7.2.2.2). We find these changes acceptable.

Changes 6, 9, and 13 are values based on revised safety analyses submitted as part of the licensee's letters. We find these changes acceptable.

Change 5 and 10 above are editorial changes resulting from the removal of the RTD bypass manifold. On the basis that these changes add clarification and conciseness to the plant's TS, we find them acceptable.

5.0 SUMMARY

The impact of the RTD bypass elimination for Millstone Unit 3, on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated and found to be acceptable. For the events impacted by the increase in the channel response time, it has been demonstrated that the conclusions presented in the FSAR remain valid. For the remaining Chapter 15 non-LOCA events, the effect of the increased initial RCS average temperature error allowance has been ascertained by separate evaluations. In all instances, the conclusions presented in the Millstone Unit 3 FSAR remain valid under this error allowance assumption and the DNBR limit value is met. The licensee's analysis to support an RCS flow measurement uncertainty value, which includes the new hot leg RTD temperature accuracy, was provided in Reference 2. This analysis was evaluated by the staff to include an RCS flow measurement uncertainty value in the Technical Specifications and was found to be acceptable after some changes were made. Because of possible degradation in RTD response time and calibration, periodic testing is needed to assure that the values assumed are consistent.

The licensee has cleaned the feedwater venturi meters used for the RCS flow measurement calibration prior to the forthcoming Cycle 2 operation. They have committed (Ref. 10) to modify the Technical Specification to indicate that a 0.1% penalty will be applied to the flow measurement uncertainty if the venturi meters are not cleaned. The staff requires that the modifications to the Technical Specifications be submitted and approved by the staff prior to Cycle 3 operation.

6.0 ENVIRONMENTAL CONSIDERATION

This 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 change surveillance requirements. The 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. 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.

7.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 20, 1988

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REFERENCES

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5. NUREG/CR-4928, Degradation of Nuclear Plant Temperature Sensors, June 1987.

6. K. R. Carr, An Evaluation of Industrial Platinum Resistance Thermometer Temperature - Its Measurement and Control in Science and Industry, ISA publication, Vol. 4, Part 2, 1972, Pages 971-982.

7. B. W. Mangum, The Stability of Small Industrial Platinum Resistance Thermometers, Journal of Research of the NBS, Vol. 89, No. 4, July-August 1984, Pages 305-350.

8. Letter from J. F. Opeka, Northeast Utility Company, to B. J. Youngblood, NRC, dated September 19, 1984.

9. ASME paper 83-JPGC-PTC-3, "Retrofitting a Flow-Section Port," C. B. Sharp, R. W. Perry.

10. Letter from E. J. Mroczka, Northeast Utilities to USNRC, dated December 23, 1987.