NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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July 5, 1985

Docket No. 50-423
B11608

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Submittal of Draft Standard Technical Specifications

Northeast Nuclear Energy Company (NNECO) hereby submits draft Standard Technical Specifications (STS) for:

1. Three loop operation of Millstone Unit No. 3.
2. Revised draft STS to reflect ANSI N510-1980.
3. Revised draft STS that provide additional information not included in the original submittal.

If you have any questions, please contact our licensing representative.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

J. F. Opeka

J. F. Opeka
Senior Vice President

B001
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STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me J. F. Opeka, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of hi^ knowledge and belief.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

ATTACHMENT

SECTION NUMBER	TITLE
2.1.1	Safety Limit - Reactor Core
2.2.1	Reactor Trip System Instrumentation Setpoints
3/4.1.1.1	Shutdown Margin - Tave 200°F
3/4.1.3.1	Group Height
3/4.1.3.2	Position Indicating System - Operating
3/4.1.3.6	Control Rod Insertion Limit
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Nuclear Enthalpy Hot Channel Factor
3/4.3.1	Reactor Trip System Instrumentation
3/4.3.3.6	Post-Accident Instrumentation
3/4.4.1.1	Reactor Coolant Loops - Startup and Power Operation
3/4.4.1.6	Isolated Loop Startup
3/4.4.5	Steam Generator
3/4.7.1.1	Tubine Cycle - Safety Valves
3/4.10.2	Group Height, Insertion and Power Distribution
3/4.3.2 (Revised)	Engineering Safety Features Actuation System Instrumentation
3/4.6.6.1 (Revised)	Supplementary Leak Collection and Release System

SECTION NUMBER	TITLE
3/4.7.7.1 (Revised)	Control Room Emergency Ventilation System
3/4.7.8 (Revised)	Auxiliary Building Filter System
3/4.9.12 (Revised)	Fuel Building Exhaust Filter System

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-67

SPECIFICATION: 2.1.1 Safety Limit - Reactor Core

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:
None.

PORC COMMENTS:

1. It was noted that this transmittal included a more legible copy of Figure 2.1-1 Reactor Core Safety Limit - Four Loops in Operation. The typographical error relating to the curves for 2250 PSIA and 2400 PSIA was corrected.

2. It was noted that the only difference from the draft specifications submitted to the NRC was the addition of Figure 2.1-2 Reactor Core Safety Limit - Three Loops in Operation as supplied by Westinghouse.

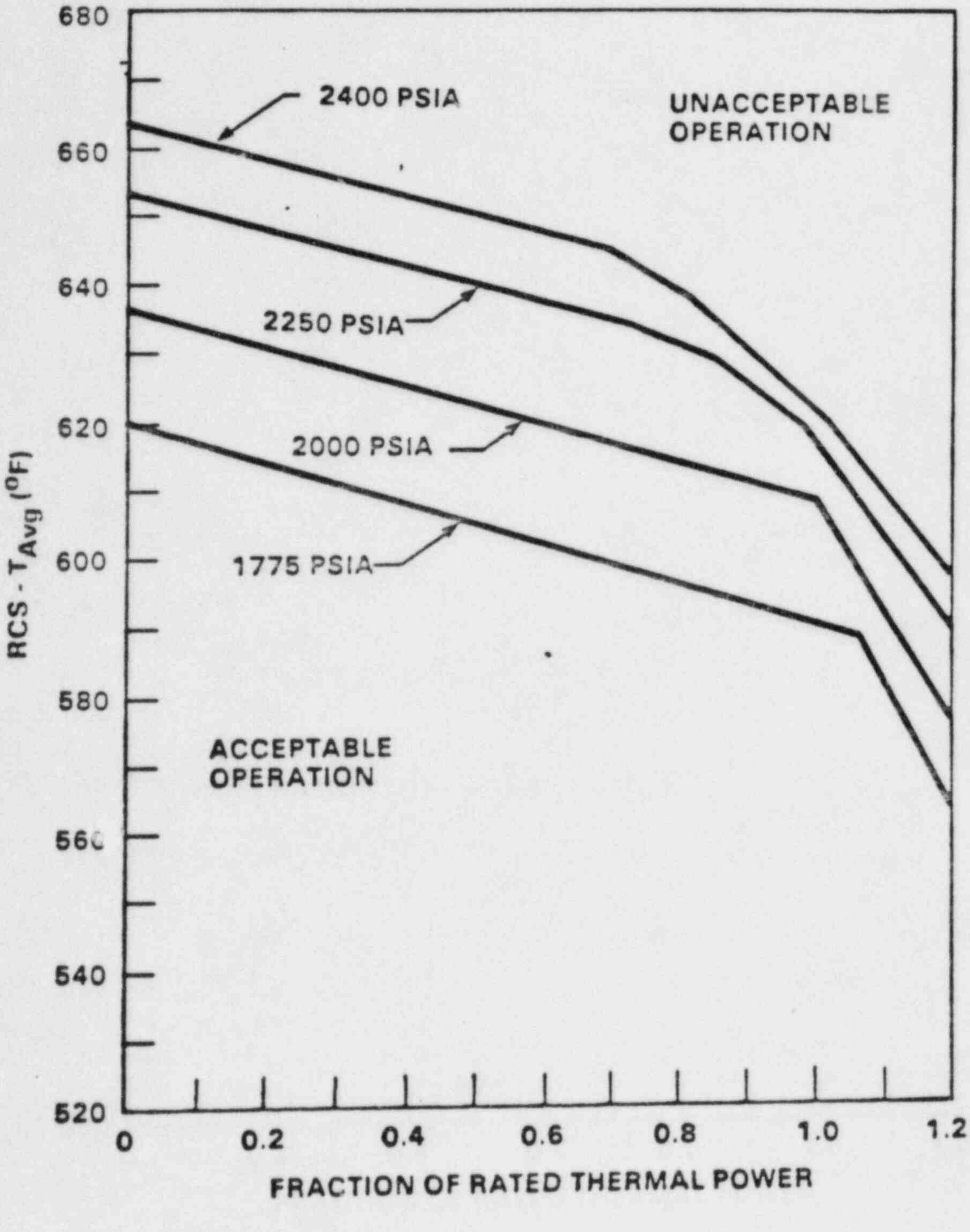

Figure 2.1-1. Reactor Core Safety Limit — Four Loops in Operation

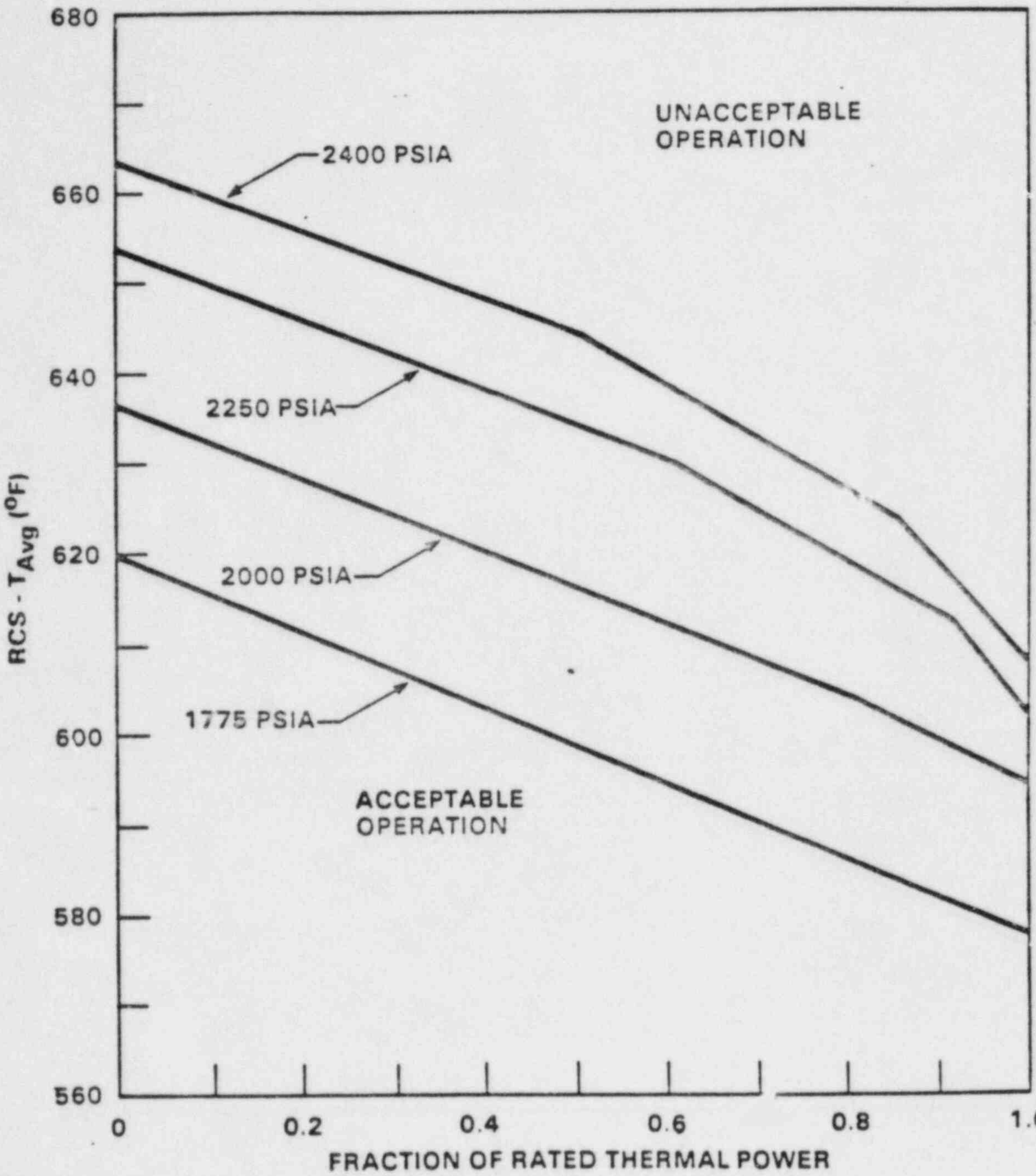

Figure 2.1-2. Reactor Core Safety Limit — Three Loops in Operation

TECHNICAL SPECIFICATION SUMMARY SHEET

SPECIFICATION: 2.1.1 Reactor Core

FSAR REFERENCE: 4.3

SER REFERENCE: ------

W STS DEVIATIONS:

1. Added plant specific safety limit curve as supplied by Westinghouse.
2. Modified bases to reflect plant design, e. g. added "R-Grid" to W-3 DNB correlation and changed "$F^N_{\Delta H} = 1.55 \{1 + 0.2(1-P)\}$ to "$F^N_{\Delta H} = 1.55 \{1 + 0.3(1-P)\}$."

SPECIFICATION: 2.1.2 RCS Pressure

FSAR REFERENCE: Table 5.2-1
Table 5.4-15

SER REFERENCE: 5.3

W STS DEVIATIONS:

1. Modified bases to reflect plant specific design.
2. Gauge pressures have been converted to absolute pressures based on a system design pressure of 2500 PSIA (2485 PSIG). This would result in an approximately 0.1% difference in hydrostatic test pressures when compared to the result obtained by adding atmospheric pressure to 125% of design gauge pressure. It is NNECO's position that this is not significant.

SPECIFICATION: 2.2.1 Limiting Safety System Settings

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Specification format was changed to include information from the statistical setpoint study.
2. RCP Shaft-Low Speed Reactor Trip has been added, RCP, UV, UF and Breaker position Trips have been deleted.
3. Delete the reference on Δ flux input to the OPΔT reactor trip.
4. ~~All values will be provided later after the final version of the statistical setpoint study is supplied by Westinghouse.~~
5. ~~Sections in the Bases on three-loop operation will be provided later.~~

4 Sections added to reflect 3 loop operation

5 Values for trip setpoints added to reflect ~~[illegible]~~ preliminary values supplied by westinghouse. Final values are still to be supplied.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

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

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

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

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

Where:

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

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

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

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

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
~~a. High Setpoint~~			~~0~~	~~≤ of RTP**~~	~~≤ of RTP**~~
INSERT A					
~~b. Low Setpoint~~			~~0~~	~~≤ of RTP**~~	~~≤ of RTP**~~
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.50	0	≤ 5.0% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.50	0	≤ 5.0% 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	≤ 1×10^5 cps	≤ 1.4×10^5 cps
INSERT B					
~~7. Overtemperature ΔT~~				~~See Note 1~~	~~See Note 2~~
8. Overpower ΔT	4.8	1.43	0.15	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.77	3.30	≥ 1900 ~~psig~~ PSIA	≥ 1890 ~~psig~~ PSIA
10. Pressurizer Pressure-High	5.0	1.77	3.30	≤ 2385 ~~psig~~ PSIA	≤ 2395 ~~psig~~ 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	1.7	0.6	≥ 90% of loop design flow*	≥ 87.2% of loop design flow*

*Loop design flow = ~~[95,700] gpm~~ 94,600 gpm (Four Loops Operating)
99,600 gpm (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
INSERT A					
2. POWER RANGE NEUTRON FLUX					
a. LOW SET POINT	8.3	4.56	0	≤ 25% of RTP**	≤ 27.1% of RTP**
b. HIGH SET POINT					
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	≤ 80.0 of RTP**	≤ 82.1 of RTP**
INSERT B					
7. OVERTEMPERATURE ΔT					
a. FOUR LOOPS OPERATING	8.3	5.9	1.6 + 1.1	See Note 1	See Note 2
b. THREE LOOPS OPERATING.	12.0	5.9	2.55	See Note 1	See Note 2

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	19.2	17.59	1.75	≥ 22.2% of narrow range instrument span	≥ 21.2% of narrow range instrument span
~~14. Steam/Feedwater Flow Mismatch Coincident With~~	INSERT C			~~≤ of full steam flow at RTP**~~	~~≤ of full steam flow at RTP**~~
~~Steam Generator Water Level Low Low~~				~~≤ of narrow range instrument span~~	~~≤ of narrow range instrument span~~
~~15. Undervoltage - Reactor Coolant Pumps~~				~~≥ volts~~	~~≥ volts~~
~~16. Underfrequency - Reactor Coolant Pumps~~		~~0~~	~~[0.1]~~	~~≥ Hz~~	~~≥ Hz~~
~~17.~~ 15. 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
~~18.~~ 16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

W-STS

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
~~19.~~ 17 Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥ 1×10^{-10} amp	≥ 6×10^{-11} amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤ 10% of RTP**	≤ 11% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Impulse Pressure Equivalent	≤ 11% RTP** Turbine Impulse Pressure Equivalent
INSERT D					
~~c. Power Range Neutron Flux, P-8~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ of RTP**~~	~~≤ of RTP**~~
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤ 50% of RTP**	≤ 53% of RTP**
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	~~≥~~ ≤ 10% of RTP**	~~≥~~ ≤ 11% of RTP**
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Impulse Pressure ~~Equivalent~~	≤ 11% ~~RTP** Turbine Impulse Pressure Equivalent~~
~~20.~~ 18 Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
~~21.~~ 19 Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

INSERT E

**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
INSERT C					
14. Reactor Coolant Pump - Low Shaft Speed	3.8	0.5	0	≥ 97.8% of Rated Speed	≥ 94.6% of Rated Speed
INSERT D					
17. C Power Range Neutron Flux, P-8					
1) Four Loops Operating	NA	NA	NA	[illegible]	([illegible])
2) Three Loops Operating	NA	NA	NA	37.7% of RTP**	39.8% of RTP.
INSERT E					
20 Three Loop Operation Bypass Circuitry	NA	NA	NA	NA	NA
21 General Warning	NA	[illegible]	[illegible]	NA	NA

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_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, τ_3 = 0 s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	*** ;
K_2	=	0.01313 /°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 33 s, τ_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, τ_6 = 0 s;

*** K_1 = 1.10 (Four Loops Operating)
1.01 (Three Loops Operating)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	≤ 587.1 °F (Nominal T_{avg} at RATED THERMAL POWER);	
K_3	= 0.000663 /psig; PSIA	
P	= Pressurizer pressure, psig; PSIA	
P'	= 2235 psig 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 1.46% 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 ** ** %.

** 2.0 % (Four Loops Operating)

** 4.1 % (Three Loops Operating)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

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_0 = As defined in Note 1,

K_4 = 1.09

K_5 = 0.0262/°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 = 10s,

$\frac{1}{1 + \tau_6 S}$ = 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 = 0.001301/°F for T > ~~T"~~ 587.1°F and K_6 = 0 for T ≤ ~~T"~~ 587.1°F,

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 3.4 %.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy and instrument drift.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels. The High Setpoint trip is reduced during three-loop operation to a value consistent with the safety analysis.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for all rod ejection ~~from mid-power.~~ accidents.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a ~~single or multiple~~ rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than ~~1.30.~~ the applicable design limit DNBR for each fuel type.

INSERT D

Overtemperature ΔT

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

INSERT E

~~OPTIONAL FOR PLANTS PERMITTED N-1 LOOP OPERATION~~

~~Operation with a reactor coolant loop out of service below the (n) loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during (n-1) loop operation exclusive of the Overtemperature delta T setpoint. (n-1) loop operation above the (n) loop P-8 setpoint is permissible after resetting the K1 input to the Overtemperature delta T channels and raising the P-8 setpoint to its (n-1) loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.~~

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no fuel pellet cracking or melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, 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, ~~(and 3) axial power distribution,)~~ 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 Break."

INSERT D

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor 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 percent 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 Protection System.

INSERT E

Three Loop Operation

Operation with a reactor coolant loop out of service requires reactor protection system modification. ~~3~~ Three loop operation is permissible after resetting the K1 input to the overtemperature delta-T channels, reducing the power range neutron flux high setpoint to a value just above the ~~3~~ three loop maximum permissible power level, and resetting the P-8 setpoint to its ~~3~~ three loop value. These modifications have been chosen so that, in three loop operation, each component of the reactor protection system performs its normal ~~4~~ four loop function, prevents operation ~~above~~ outside the safety limit curves, and prevents the DNBR from going below 1.30 during normal operational and anticipated transients.

Pressurizer Pressure

In each of the 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 percent of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10 percent 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 high water level 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 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full equivalent); and on increasing power, automatically reinstated by P-7.

Loss of Flow

The ~~Loss of Flow~~ Reactor Coolant Flow Low trip~~s~~ provide 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 percent of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 percent 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 ~~30~~ percent of RATED THERMAL POWER~~)~~ (for four loop operation, and percent of RATED THERMAL POWER for three loop operation) an automatic reactor trip will occur if the flow in any single loop drops below (90 percent) of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Loss of Flow (Continued)

~~OPTIONAL FOR PLANTS PERMITTED N-1 LOOP OPERATION~~

~~The P-8 setpoint trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when (n-1) loops are in operation and the Overtemperature delta T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip setpoint adjusted to the value specified for (n-1) loop operation, the P-8 trip at (76%) RATED THERMAL POWER will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with (n-1) loops in operation.~~

Steam Generator Water Level

The steam generator water level low-low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the auxiliary feedwater system.

~~Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level~~

~~The steam/feedwater flow mismatch in coincidence with a steam generator~~ low water level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to (1.42×10^6) lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below (25) percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide reactor core protection against DNB as a result of complete loss of forced coolant flow. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious ~~reactor trips from momentary electrical power transients. For undervoltage,~~

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~~Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)~~

~~the delay is set so that the time required for a signal to reach the reactor~~ trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed (1.2) seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed (0.3) seconds. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on ~~increasing power, reinstated automatically by P-7.~~

Turbine Trip

A Turbine Trip initiates a reactor trip. On decreasing power the turbine trip is automatically blocked by P-~~7~~ 9 (a power level of approximately ~~10~~ 50 percent of RATED THERMAL POWER ~~with a turbine impulse chamber at approximately 10 percent of full power equivalent~~); and on increasing power, reinstated automatically by P-~~7~~ 9.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

~~Reactor Coolant Pump Breaker Position Trip~~ INSERT F

~~The Reactor Coolant Pump Breaker Position Trips are anticipatory trips~~ which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System. Above P-7 (a power level of approximately 10 percent of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 percent of full power equivalent) an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 30 percent of RATER THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7 an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is ~~automatically blocked.~~

INSERT F

Low Shaft Speed
~~Underspeed~~ - Reactor Coolant Pumps

RCP - Low Shaft Speed
The ~~Underspeed~~ ~~- Reactor Coolant Pumps~~ trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on ~~all four reactor coolant pumps.~~ two reactor coolant pumps in any two operating reactor coolant loops. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above 1.30 following a 4 pump loss of flow event.

Three Loop Operation Bypass Circuitry

The three loop operation bypass circuitry reactor trip insures that a sufficient number and the correct combinations of protection circuit remain available to provide necessary protection and mitigation capability during three loop operation. Should more than two channels in one train or two dissimilar channels in two trains be bypassed, a reactor trip will occur. In this manner it is insured that sufficient protective features remain to mitigate the consequences of analyzed transient.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables reactor trips on low flow in more than one primary coolant loop, reactor coolant pump low shaft speed, ~~more thn one reactor coolant pump breaker open; reactor coolant pump bus undervoltage and underfrequency~~, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked.

P-8 On increasing power P-8 automatically enables reactor trips on low flow in one or more primary coolant loops. ~~and one or more reactor coolant pump breakers open.~~ On decreasing power the P-8 automatically blocks the above listed trip~~s~~.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the flow setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. On decreasing power the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

P-9 On increasing power, P-9 automatically enables the reactor trip from turbine trip. On decreasing power, the above trip is automatically blocked

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-68

SPECIFICATION: 3/4.1.1.1 Shutdown Margin Tave $>$200°F

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:
None.

PORC COMMENTS:

1. It was noted that the only difference from the draft specifications submitted to the NRC was the deletion of the reference to N-loop operation. The SHUTDOWN MARGIN for both 3 and 4 loop is bounded by the 4 loop value.

SPECIFICATION: 3/4.1.1.4 Minimum Temperature for Criticality

FSAR REFERENCE: ------

SER REFERENCE: ------

[Delete [illegible] Draft 1 [illegible]
[illegible] up]

W STS DEVIATIONS:
None.

PORC COMMENTS:

1. It was noted that the only difference from the draft specifications submitted to the NRC was the incorporation of the Unit 3 minimum temperature into the action statement.

SPECIFICATION: 3/4.1.3 Movable Control Assemblies - Group Height

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Deleted the standard, which had been submitted to the NRC, and replaced it with a specification that deviates from the standard as follows:

   a. Split old action c.3.d) into two actions b.3.c) and b.3.d) to reflect 3 loop operating requirements.

   ~~b. Deleted old action c.3.b). This is a duplication of surveillance 4.1.1.1.1.a.~~

c. Split old action b. into two actions c. and d. This would allow additional flexibility for the case where more than one rod is not operable but still trippable. The new action requirement is consistent with old action c.2. For the case where more than one rod is misaligned, the action requirements have not changed.

PORC COMMENTS:

1. Modified LCO such that all rods are included, not just rods inserted in the core. This is consistent with the draft specification submitted to the NRC.

SPECIFICATION: 3/4.1.3.2 Position Indication Systems - Operating

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. It was noted that the only difference from the draft specifications submitted to the NRC was the modification of actions a.2 and b.2 to reflect 3 loop operating requirements.

PORC COMMENTS:

1. PORC directed that the wording from draft Revision 5 of the STS be used for this specification. These changes reflect the Digital Rod Position Indication system used at Unit 3.

SPECIFICATION: 3/4.1.3.4 Rod Drop Time

Delete, not submitted

This specification has not been changed from that which was submitted to the NRC. It was included for review pending the incorporation of missing information. The information was not received; thus, this specification will be held in abeyance.

SPECIFICATION: 3/4.1.3.6 Control Rod Insertion Limits

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

None.

PORC COMMENTS:

1. It was noted that the only difference from the draft specifications submitted to the NRC was the addition of Figure 3.1-2, Rod Bank Insertion Limits versus Thermal Power - Three Loop Operation.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

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LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to [1.6%] Δk/k. ~~for [n] loop operation.~~

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

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

4.1.1.1.1 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 above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

~~b. With more than one full-length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.~~

b ~~c~~. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

1. The rod is restored to OPERABLE status within the above alignment requirements, or
2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or
3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:
   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;
   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

Four Loops Operating

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

INSERT

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

INSERT

e) Three Loops Operating
The THERMAL POWER level is reduced to less than or equal to 50% of RATED THERMAL POWER within the next hour and within the following 4 hours the neutron flux high trip setpoint is reduced to less than or equal to 60% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a above, POWER OPERATION may continue provided that:

1. Within one hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from it's group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

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POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps. D

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod pos[illegible]dicator per bank inoperable either: C

1. Determine the position of the non[illegible]dicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

~~2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours~~.

INSERT A

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or C

~~2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours~~.

INSERT B

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours. D + C

INSERT A

2 Four Loops Operating
Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours

3. Three Loops Operating
Reduce THERMAL POWER to less than 32% of RATED THERMAL POWER within 8 hours.

INSERT B

2. Four Loops Operating
Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

3. Three Loops Operating
Reduce THERMAL POWER to less than 32% of RATED THERMAL POWER within 8 hours.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

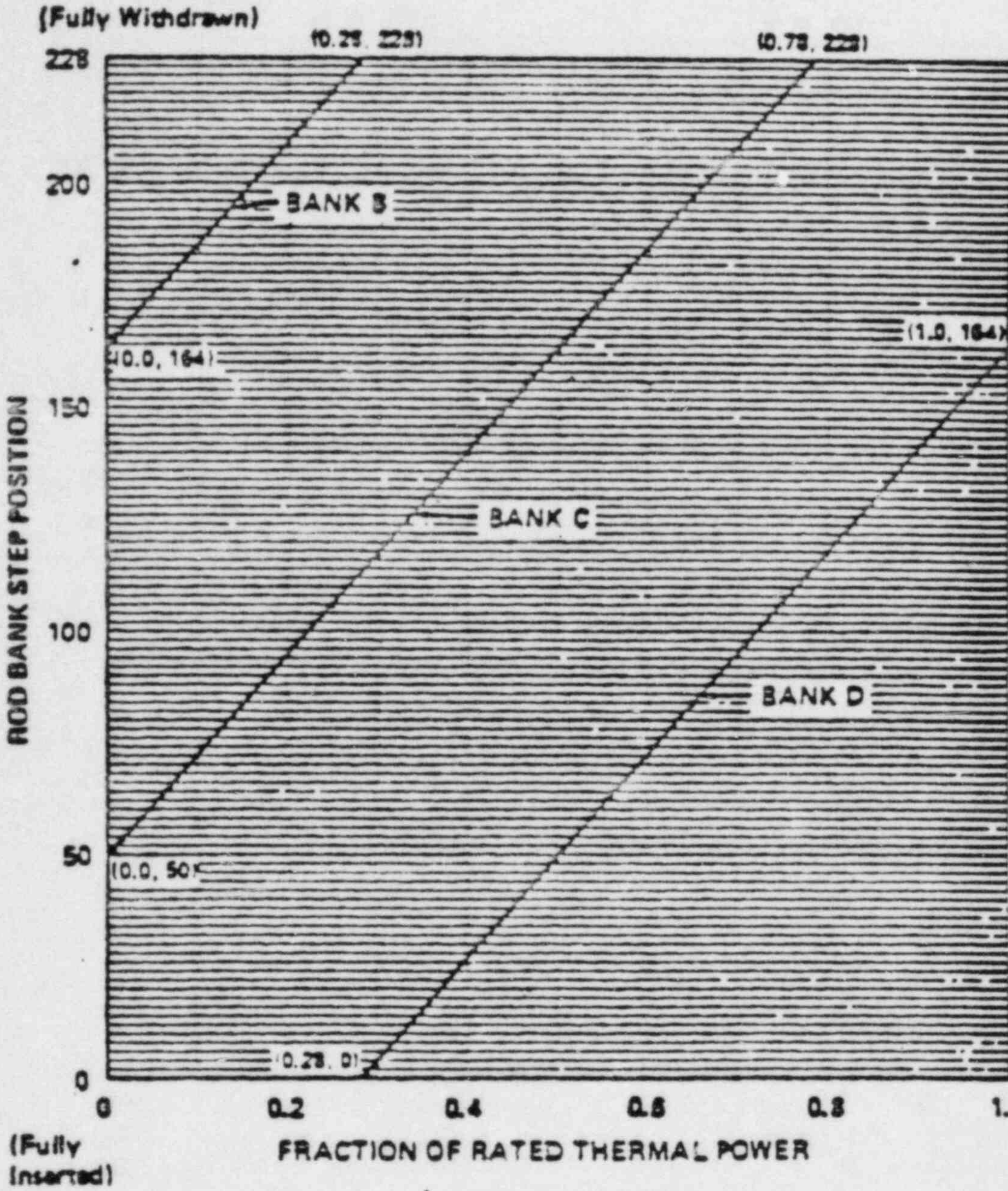

FIGURE ~~2.1-2~~ 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

Figure I

~~NEW N-1 Loop RIL Curve~~

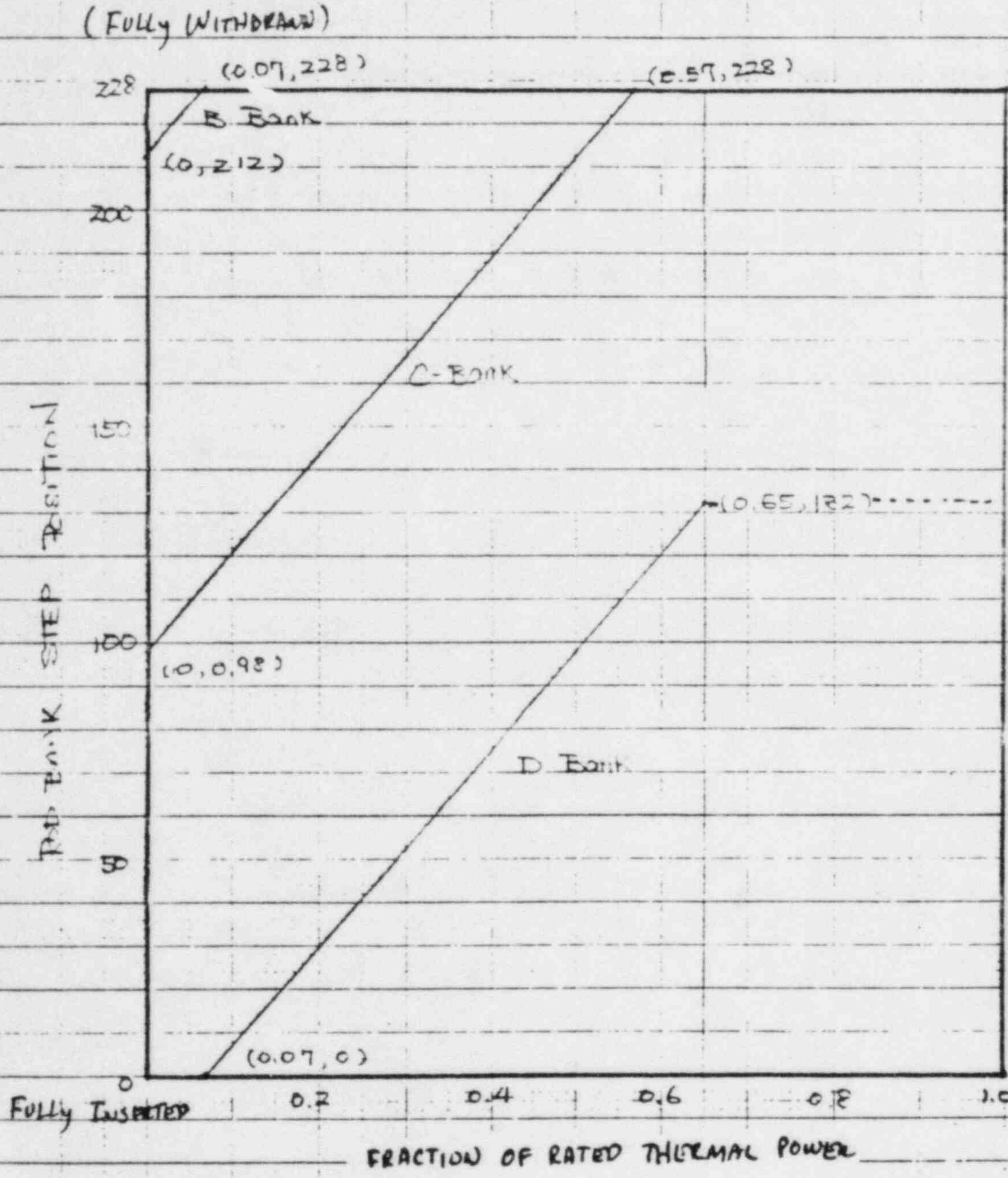

FIGURE 3.1-2

ROD BANK INSERTION LIMITS
VERSUS THERMAL POWER
THREE LOOP OPERATION

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3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of [1.6%] $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 2% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection. In addition, the Mode 5 SHUTDOWN MARGIN is consistent with the assumptions used in the boron dilution analysis.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

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BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.6% Δk/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 [5106] gallons of 6300 ppm borated water from the boric acid storage tanks or [52,622] gallons of 2000 ppm borated water from the refueling water storage tank (RWST). To be consistent with specification 3/4.5.4 a minimum RWST volume of 1,166,000 gallons is specified.

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 [275]°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 2% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either ___ gallons of 6300 ppm borated water from the boric acid storage tanks or ____ gallons of 2000 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 freeze 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.

INSERT C

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

INSERT A

The minimum RWST solution temperature for Modes 5 and 6 is based on freeze considerations of the tank contents. The maximum/minimum solution temperatures for Modes 1, 2, 3 and 4 are based on analysis assumptions.

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MOVABLE CONTROL ASSEMBLIES (Continued)

rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with ~~all~~ the specified number of reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

add

For Specification 3.1.3.1 ACTIONS b and c it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus fall under the requirements of ACTION A. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately four hours for this verification.

Ref. TS-12

TECHNICAL SPECIFICATION
SUMMARY SHEET

SPECIFICATION: 3/4.2.1.1 and 3/4.2.1.2 Axial Flux Difference

FSAR REFERENCE: 4.3.2.2.6

SER REFERENCE: 4.3.2

W STS DEVIATION:

1. Changed the Bases to "required by Specification 3.2.1" rather than "specify ±5%."
2. Added section to reflect Three loop operation

SPECIFICATION: 3/4.2.2.1 and 3/4.2.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$

FSAR REFERENCE: 4.3.2.2.6

SER REFERENCE: 4.3.2
4.4.4.1

W STS DEVIATIONS:

1. Deleted references to APDMS as Millstone 3 is not required to have this equipment.
2. Deleted 17x17 fuel elements as we do not need to specify fuel type here.
3. Deleted references to part length rods.
4. Changes ±13 in the Bases to ±12 to be consistent with the rest of the Technical Specifications.
5. Added section to reflect 3 loop operation

SPECIFICATION: 3/4.2.3.1 and 3/4.2.3.2 RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor

FSAR REFERENCE: 4.3.2.2.6
4.4.2.2.6
4.4.4.3

SER REFERENCE: 4.4.4.1

W STS DEVIATIONS:

1. Deleted references to R_2 and the ROD BOW PENALTY to be consistent with the FSAR and incorporate recommendations made by Westinghouse.
2. ~~2.~~ The 2.1% flow uncertainty for flow in Figure 3.2-3 may change as a result of analysis being done in response to NRC Question 492.7.

2 Deleted Figure 3.2-3 and R factor to make the specification read $F^N_{\Delta H} \leq$ and RCS flow $\geq$

3 Added section to reflect 3 Loop operation

4. Flow uncertainty changed to 2.4% to reflect MP-3 specific instrumentation

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

Four Loops Operating

LIMITING CONDITION FOR OPERATION

3.2.1.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU; and
b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1a and the cumulated penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

  1. Restore the indicated AFD to within the above required target band limits, or

  2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure ~~3.2-1~~ 3.2-1a and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

  1. THERMAL POWER to less than 50% of RATE THERMAL POWER within 30 minutes, and

  2. The Power Range Neutron Flux-High# * Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

*See Special Test Exception Specification 3.10.2.

#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification ~~4.3.1.1~~ 4.3.1.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1a. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.1.2 ~~4.2.1.2~~ The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.1.3 ~~4.2.1.3~~ The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.1.4 ~~4.2.1.4~~ The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification ~~4.2.1.3~~ above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

4.2.1.1.3

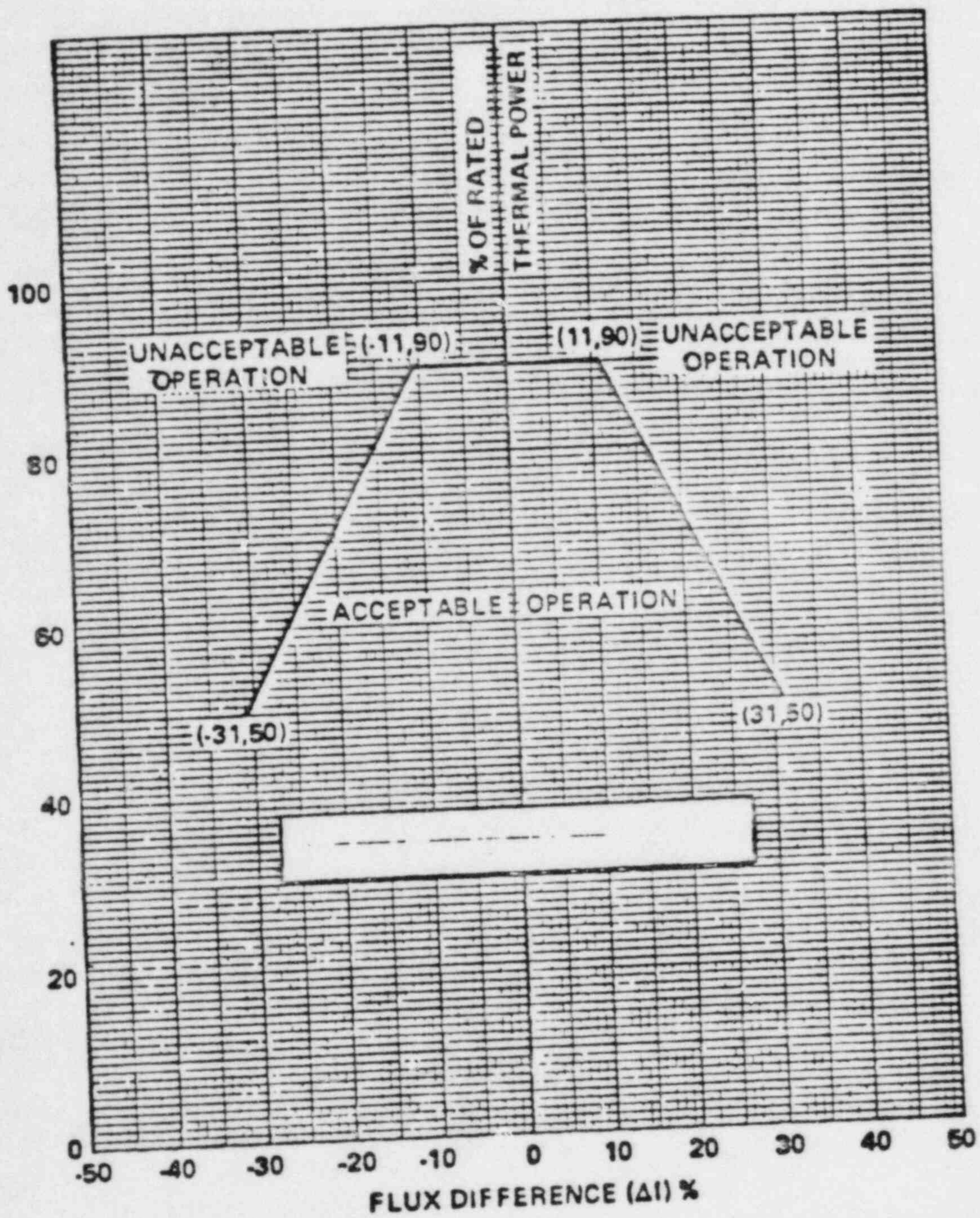


FIGURE 3.2-1a AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

Four Loops Operating

3/4.2 POWER DISTRIBUTION LIMITS

~~3/4.2.1 AXIAL FLUX DIFFERENCE~~

Three Loops Operation

LIMITING CONDITION FOR OPERATION

3.2.1.2 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within ±5% of the ~~following~~ target band (flux difference units) about the target flux difference~~:~~.

~~a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU; and~~

~~b. + 3%, - 12% for core average accumulated burnup of greater than 3000 MWD/MTU~~.

The indicated AFD may deviate outside the above required target band at greater than or equal to ~~50%~~ 32% but less than ~~90%~~ 65% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1b and the cumulated penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than ~~50%~~ 32% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to ~~90%~~ 65% of RATED THERMAL POWER, within 15 minutes, either:

   1. Restore the indicated AFD to within the above required target band limits, or

   2. Reduce THERMAL POWER to less than ~~90%~~ 65% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure ~~3.2-1~~ 3.2-1b and with THERMAL POWER less than ~~90%~~ 65% but equal to or greater than ~~50%~~ 32% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than ~~50%~~ 32% of RATE THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux-High# Setpoints to less than or equal to ~~55%~~ 37% of RATED THERMAL POWER within the next 4 hours.

*See Special Test Exception Specification 3.10.2.

#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification ~~4.3.1.1~~ 4.3.1.2.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure ~~3.2-1~~ 3.2-1b. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than ~~50%~~ 32% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than ~~50%~~ 32% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.2.1 ~~4.2.1.1~~ The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2.2 ~~4.2.1.2~~ The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

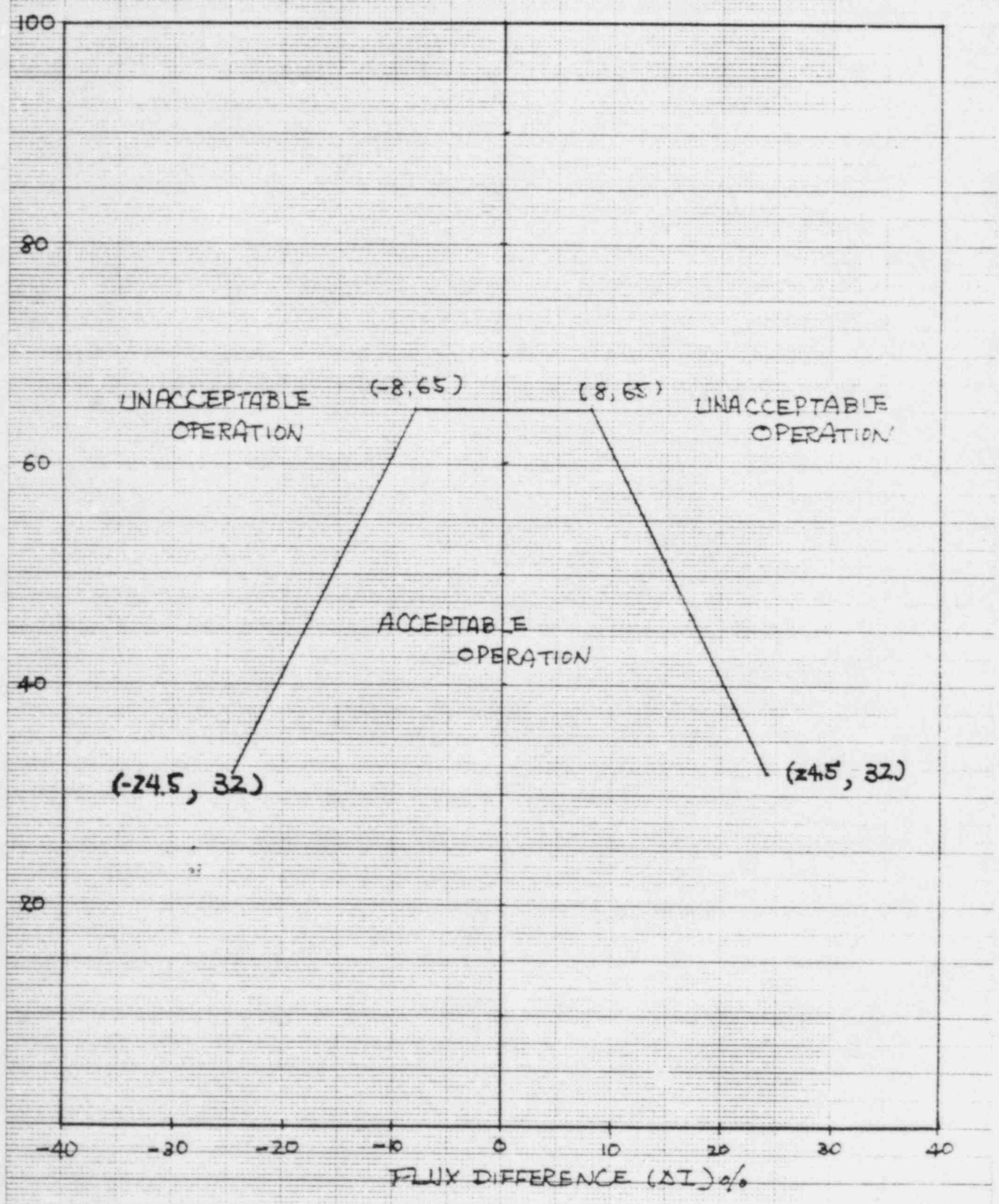
a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above ~~50%~~ 32% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and ~~50%~~ 32% of RATED THERMAL POWER.

4.2.1.2.3 ~~4.2.1.3~~ The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2.4 ~~4.2.1.4~~ The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification ~~4.2.1.3~~ 4.2.1.2.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

FIG 3.2-1b


100
80
60
40
20
UNACCEPTABLE
OPERATION
(-8, 65)
(8, 65)
UNACCEPTABLE
OPERATION
ACCEPTABLE
OPERATION
(-24.5, 32)
(24.5, 32)
-40
-30
-20
-10
0
10
20
30
40
FLUX DIFFERENCE (ΔI) %

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

Four Loops Operating

LIMITING CONDITION FOR OPERATION

3.2.2.1 ~~3.2.2~~ $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq [\frac{2.32}{P}] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-22 ~~(3.2-2)~~ for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

~~a. Comply with either of the following ACTIONS:~~

~~1.~~ Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. ~~The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.~~

~~2. Reduce THERMAL POWER as necessary to meet the limits of Specification (3.2.6) using the APDMS with the latest incore map and updated R. (APDMS plants only)~~

~~b.~~ Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit. ~~required by a., above;~~ THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1.1
~~4.2.2.1~~ The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2
~~4.2.2.2~~ F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

   1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

   2. The relationship:

   $$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)]$$

   where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

   1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

      a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

      b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification ~~6.9.1.10.~~ 6.9.1.9

f. The F_{xy} limits of e, above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at ~~17.8~~ 18.1 ± 2%, ~~32.1~~ 32.3 ± 2%, ~~46.4~~ 46.6 ± 2%, ~~60.6~~ 60.9 ± 2% and ~~74.9~~ 75.1 ± 2%, inclusive ~~(17 x 17 fuel elements)~~.

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" ~~or part-length~~ control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

~~1. The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}, and (for plants with $F_Q(Z)$ less than 2.32 and using APDMS)~~

~~2.~~ ~~T~~he effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

~~4.2.2.3~~ 4.2.2.1.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

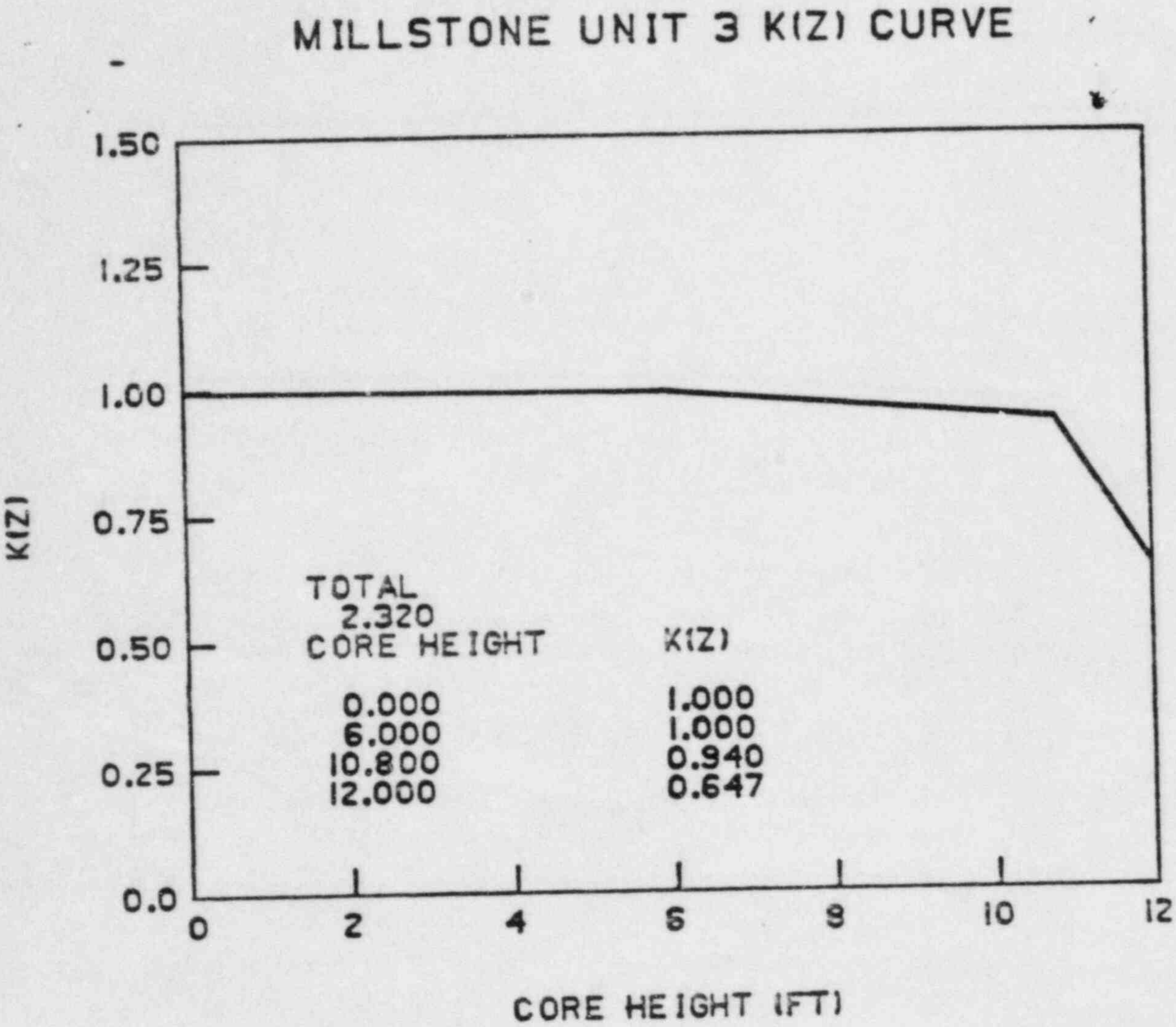

MILLSTONE UNIT 3 K(Z) CURVE
1.50
1.25
1.00
0.75
0.50
0.25
0.0
K(Z)
TOTAL
2.320
CORE HEIGHT
0.000
6.000
10.800
12.000
K(Z)
1.000
1.000
0.940
0.647
0
2
4
6
8
10
12
CORE HEIGHT (FT)

Figure 3.2-2

~~3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$~~

Three Loops Operating

LIMITING CONDITION FOR OPERATION

3.2.2.2 ~~3.2.2~~ $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{\cancel{2.32}\ 2.60}{P}\right] [K(Z)] \text{ for } P > \cancel{0.5}\ 0.375$$

$$F_Q(Z) \leq [\cancel{4.64}\ 5.20] [K(Z)] \text{ for } P \leq \cancel{0.5}\ 0.375$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure ~~(3.2-2)~~ 3.2-2b for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

~~a. Comply with either of the following ACTIONS:~~

~~1.~~ a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similiarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

~~2. Reduce THERMAL POWER as necessary to meet the limits of Specification (3.2.6) using the APDMS with the latest incore map and updated R. (APDMS plants only)~~

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit ~~required by a, above~~. THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 ~~4.2.2.1~~ The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 ~~4.2.2.2~~ F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

65% of

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

$F_{xy}^{0.65 RTP}$

2. The relationship:

$F_{xy}^{0.65 RTP}$ $M F_{xy}$

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

$F_{xy}^{0.65 RTP}$, MF_{xy} is the F_{xy} multiplier,

d. Remeasuring F_{xy} according to the following schedule:

.65 RTP

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} .65 RTP and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the ~~F_{xy}^{RTP}~~ $F_{xy}^{0.65RTP}$ limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to ~~F_{xy}^{RTP}~~ $F_{xy}^{0.65RTP}$ and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for 65% of RATED THERMAL POWER (~~F_{xy}^{RTP}~~ $F_{xy}^{0.65RTP}$) and M_{xy} multiplier ($M_{F_{xy}}$) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification ~~6.9.1.10~~ 6.9.1.7.

f. The F_{xy} limits of e, above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at ~~17.8~~ 18.1 ± 2%, ~~32.1~~ 32.3 ± 2%, ~~46.4~~ 46.6 ± 2%, ~~60.6~~ 60.9 ± 2% and ~~74.9~~ 75.1 ± 2%, inclusive ~~(17 x 17 fuel elements)~~.

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" ~~or part-length~~ control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L},

~~1. The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}, and (for plants with $F_Q(Z)$ less than 2.32 and using APDMS)~~

~~2.~~ The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

~~4.2.2.3~~ 4.2.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

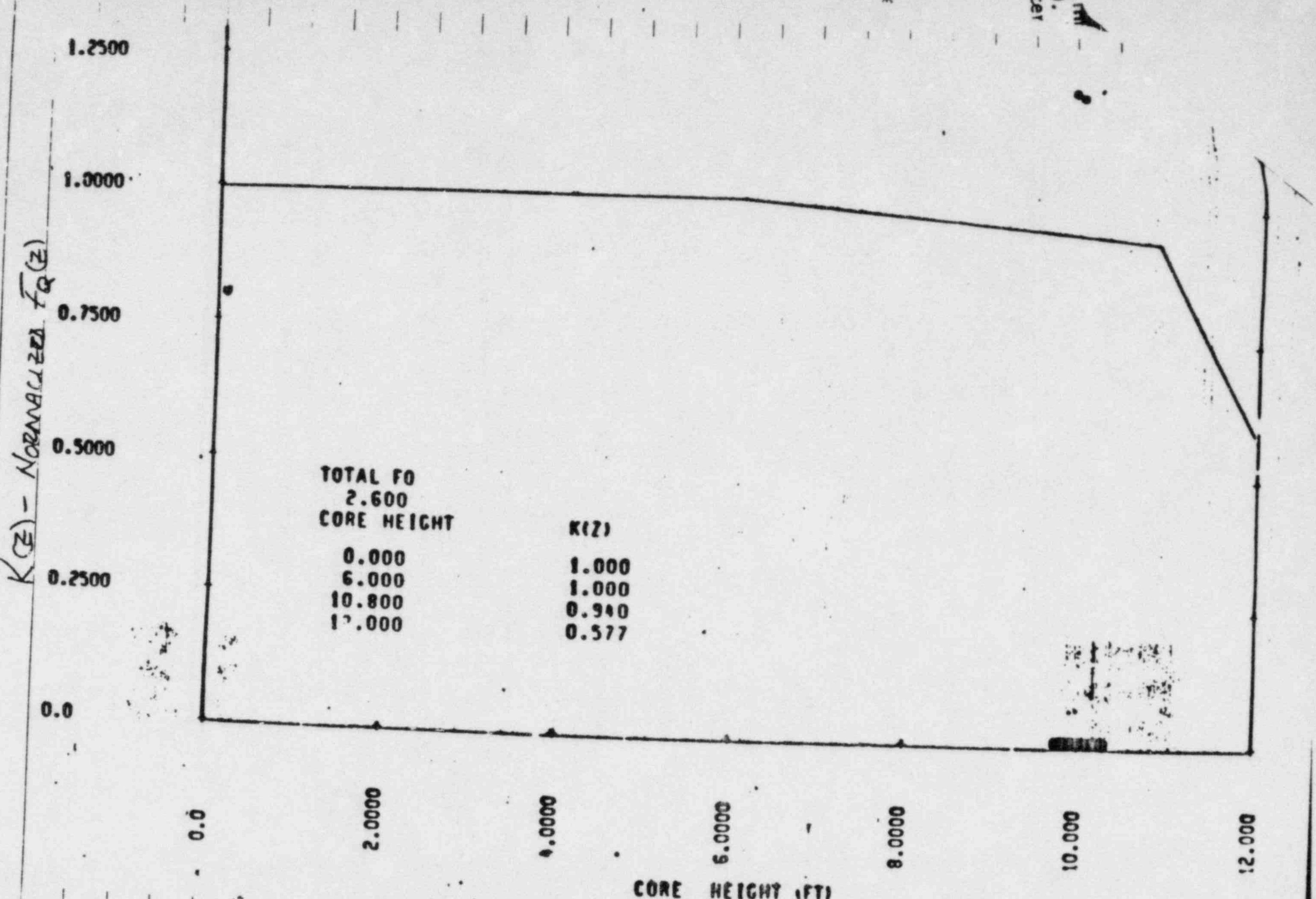


FIGURE 3.2-2.6 K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT FOR 3 LOOP OPERATION

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

Four Loops Operating

LIMITING CONDITION FOR OPERATION

3.2.3.1 The ~~combination~~ $F^N_{\Delta H}$ of indicated Reactor Coolant System (RCS) total flow rate and ~~R_1, R_2~~ shall be maintained ~~within the region of allowable operation shown on Figure 3.2-3 for 4 loop operation.~~ as follows:

INSERT A

~~Where:~~

~~a. $R_1 = \frac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$,~~

~~b. $R_2 = \frac{R_1}{[1-RBP(BU)]}$,~~

a. $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

b. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. ~~The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$, and~~ ~~have been included in the~~

INSERT B

~~e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).~~

APPLICABILITY: MODE 1.

ACTION:

With the ~~combination of~~ RCS total flow rate ~~and R_1, R_2~~ or $F^N_{\Delta H}$ outside the region of acceptable operation; ~~shown on Figure 3.2-3~~:

a. Within 2 hours either:

1. Restore the ~~combination of~~ RCS total flow rate and ~~R_1, R_2~~ $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWFR to less than 50% of RATED THERMAL POWER and reduce the Powe, Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

INSERT A

a. RCS total flow $\geq$ 387,500 gpm, and

b. $F^{N}_{\Delta H} \leq 1.49\,[1.0 + 0.3\,(1.0 - P)]$.

INSERT B

c. The measured value of RCS total flow rate shall be used since uncertainties of 2.4 % for flow measurement have been included in the LCO.

POWER DISTRIBUTION LIMITS

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate ~~comparison~~ that ~~the combination of R_1, R_2~~ $F^N_{\Delta H}$ 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 items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that ~~the combination of R_1, R_2~~ $F^N_{\Delta H}$ 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 ~~shown on Figure 3.2-3~~ 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 ~~4.2.3.1~~ The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 ~~4.2.3.2~~ ~~The combination of indicated~~ RCS total flow rate ~~and R~~ $F^N_{\Delta H}$ shall be determined to be within ~~the region of~~ acceptable ~~operation of Figure 3.2-3~~ 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 ~~4.2.3.3~~ The indicated range RCS total flow rate shall be verified to be within the ~~region of~~ acceptable ~~operation of Figure 3.2-3~~ at least once per 12 hours when the most recently obtained value of ~~R,~~ $F^N_{\Delta H}$ obtained per Specification ~~4.2.3.2,~~ 4.2.3.1.2 is assumed to exist.

4.2.3.1.4 ~~4.2.3.4~~ The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.1.5 ~~4.2.3.5~~ 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 ~~4.2.3.6~~ The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months. If venturi is not determined clean the minimum required RCS flow of 3.2.3. shall be increased by .1%

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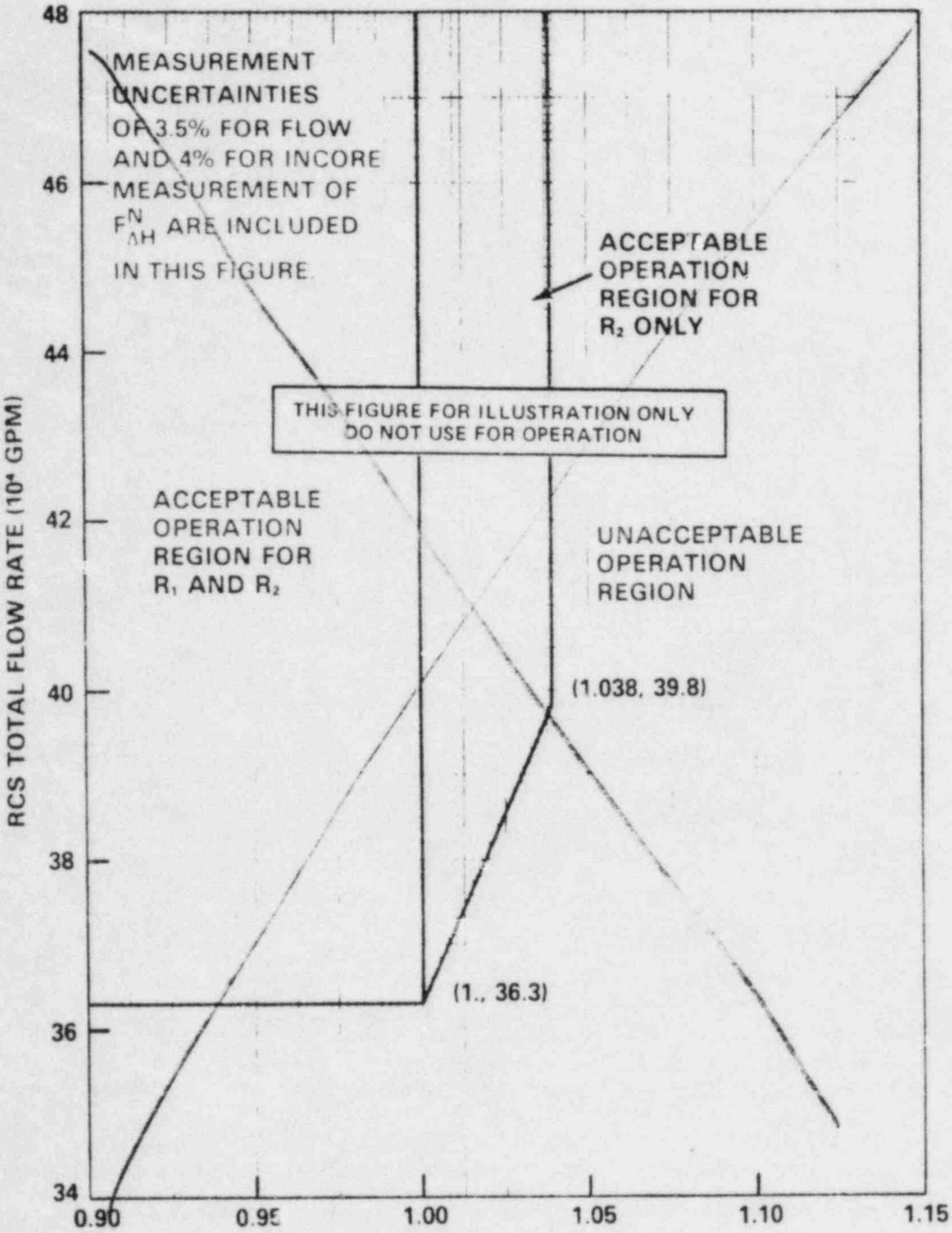


$R_1 = F^N_{\Delta H}/1.49\ [1.0 + 0.2\ (1.0 - P)]$

$R_2 = R_1/[1 - RBP(BU)]$

FIGURE 3.2-3 RCS TOTAL FLOWRATE VERSUS R — FOUR LOOPS IN OPERATION

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

Three Loops Operating

LIMITING CONDITION FOR OPERATION

3.2.3.2 ~~3.2.3~~ The ~~combination of~~ indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ ~~R_1, R_2~~ shall be maintained ~~within the region of allowable operation shown on Figure 3.2-3 for 4 loop operation.~~ as follows:

INSERT C

~~Where:~~

~~a. $R_1 = \frac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$,~~

~~b. $R_2 = \frac{R_1}{[1 - RBP(BU)]}$,~~

a. $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, $\geq 65\%$ of

b. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. ~~The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3. includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$ and~~ [illegible]

INSERT D

~~e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).~~

APPLICABILITY: MODE 1.

ACTION:

With the ~~combination of~~ RCS total flow rate ~~and R_1, R_2~~ or $F^N_{\Delta H}$ outside the ~~region of~~ acceptable ~~operation shown on Figure 3.2-3.~~ range

a. Within 2 hours either:

1. Restore the ~~combination of~~ RCS total flow rate ~~and R_1, R_2~~ and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than ~~60%~~ 32% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to ~~55%~~ 37% of RATED THERMAL POWER within the next 4 hours.

INSERT C

a. RCS total flow $\geq$ 305,000 gpm, and

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

INSERT D

c. The measured value of RCS total flow rate shall be used since uncertainties of 2.4 % for flow measurement have been included in the LCO.

ACTION: (Continued)

indicated

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate ~~comparison~~ that ~~the combination of R_1,~~ R_2 $F_{\Delta H}^N$ 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 items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that ~~the combination of R_1, R_2~~ $F_{\Delta H}^N$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate ~~comparison~~, to be within the ~~region~~ range of acceptable ~~operation shown on Figure 3.2-3~~ prior to exceeding the following THERMAL POWER levels:

1. A nominal ~~50%~~ 32% of RATED THERMAL POWER,
2. A nominal ~~75%~~ 50% of RATED THERMAL POWER, and
3. ~~Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.~~

SURVEILLANCE REQUIREMENTS

4.2.3.2.1 ~~4.2.3.1~~ The provisions of Specification 4.0.4 are not applicable.

4.2.3.2.2 ~~4.2.3.2~~ The ~~combination of~~ indicated RCS total flow rate and ~~R~~ $F_{\Delta H}^N$ shall be determined to be within ~~the region of~~ acceptable ~~operation of Figure 3.2-3~~ 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.

4.2.3.2.3 ~~4.2.3.3~~ The indicated RCS range total flow rate shall be verified to be within the ~~region of~~ acceptable ~~operation of Figure 3.2-3~~ at least once per 12 hours when the most recently obtained value of ~~R~~ $F_{\Delta H}^N$, obtained per Specification ~~4.2.3.2~~ 4.2.3.2.2, is assumed to exist.

4.2.3.2.4 ~~4.2.3.4~~ The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.2.5 ~~4.2.3.5~~ 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.

4.2.3.2.6 ~~4.2.3.6~~ The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months. If venturis are not determined clean the minimum required RCS flow of 3.2.3.2 shall be increased by .1%

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (b) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

2.32 (four loops operating) or 2.60 (three loops operating)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.32~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

shutdown and contro

Target flux difference is determined at equilibrium xenon conditions ~~with the part-length control rods withdrawn from the core.~~ The ~~full-length~~ rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued) (required by Specification 3.2.1

Although it is intended that the plant will be operated with the AFD within the ~~±5%~~ target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band ~~but within the limits of Figure (3.2-1) while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER~~, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance. INSERT E

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for 2 or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively, when in four loop operation

INSERT F

Figure ~~B 3/4 2-1~~ B 3/4 2-1a and B 3/4 2-1b shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± ~~15~~ 12 steps, indicated, from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

Insert E

... but within the limits of Figure 3.2-12 and 3.2-16 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER (four loops operating) or between 32% and 65% of RATED THERMAL POWER (three loops operating). For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER (four loops operating) or between 15% and 32% of RATED THERMAL POWER (three loops operating), ...

Insert F

During three loop operation at THERMAL POWER level between 32% and 65% and between 15% and 32% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively

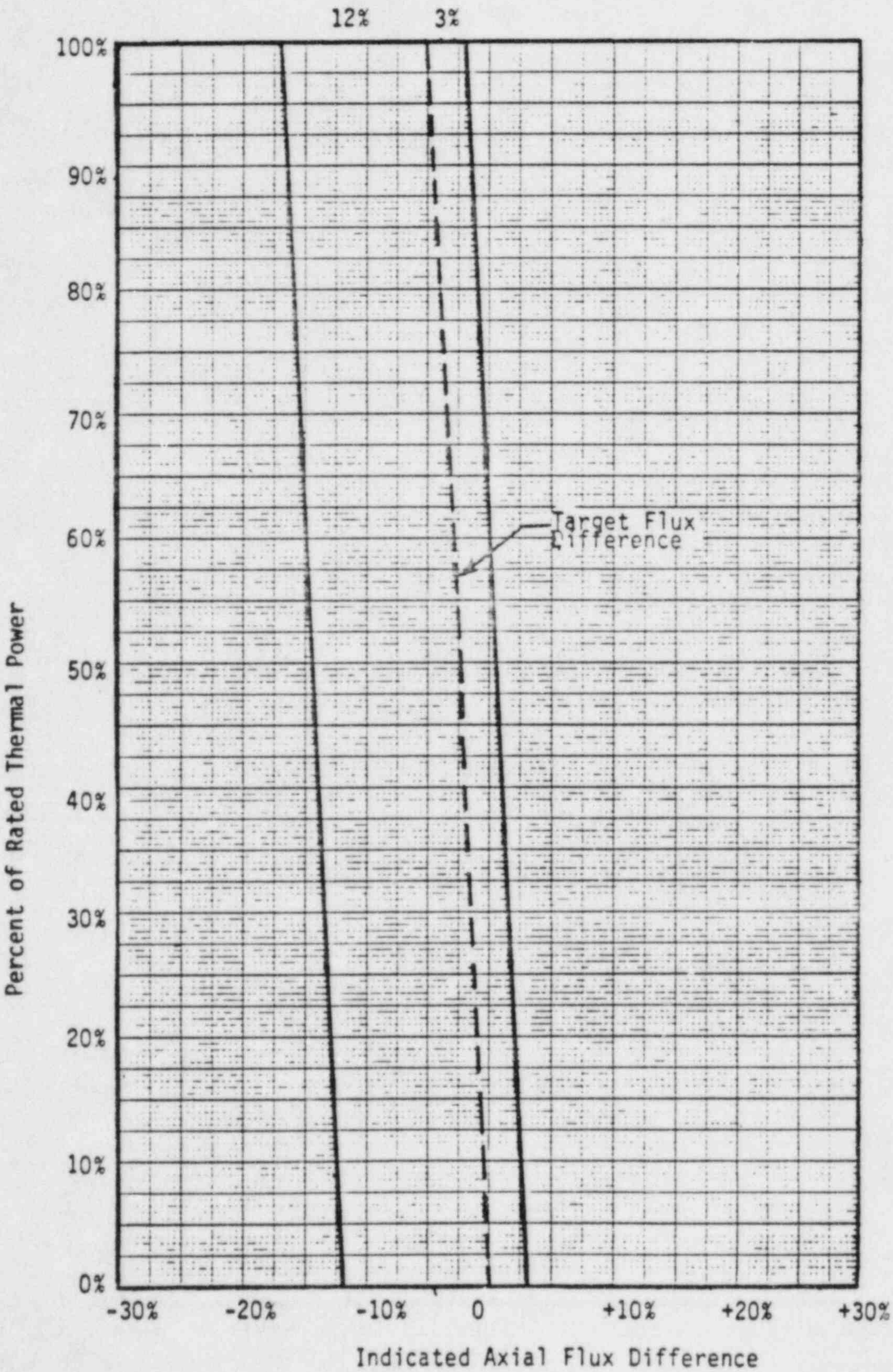
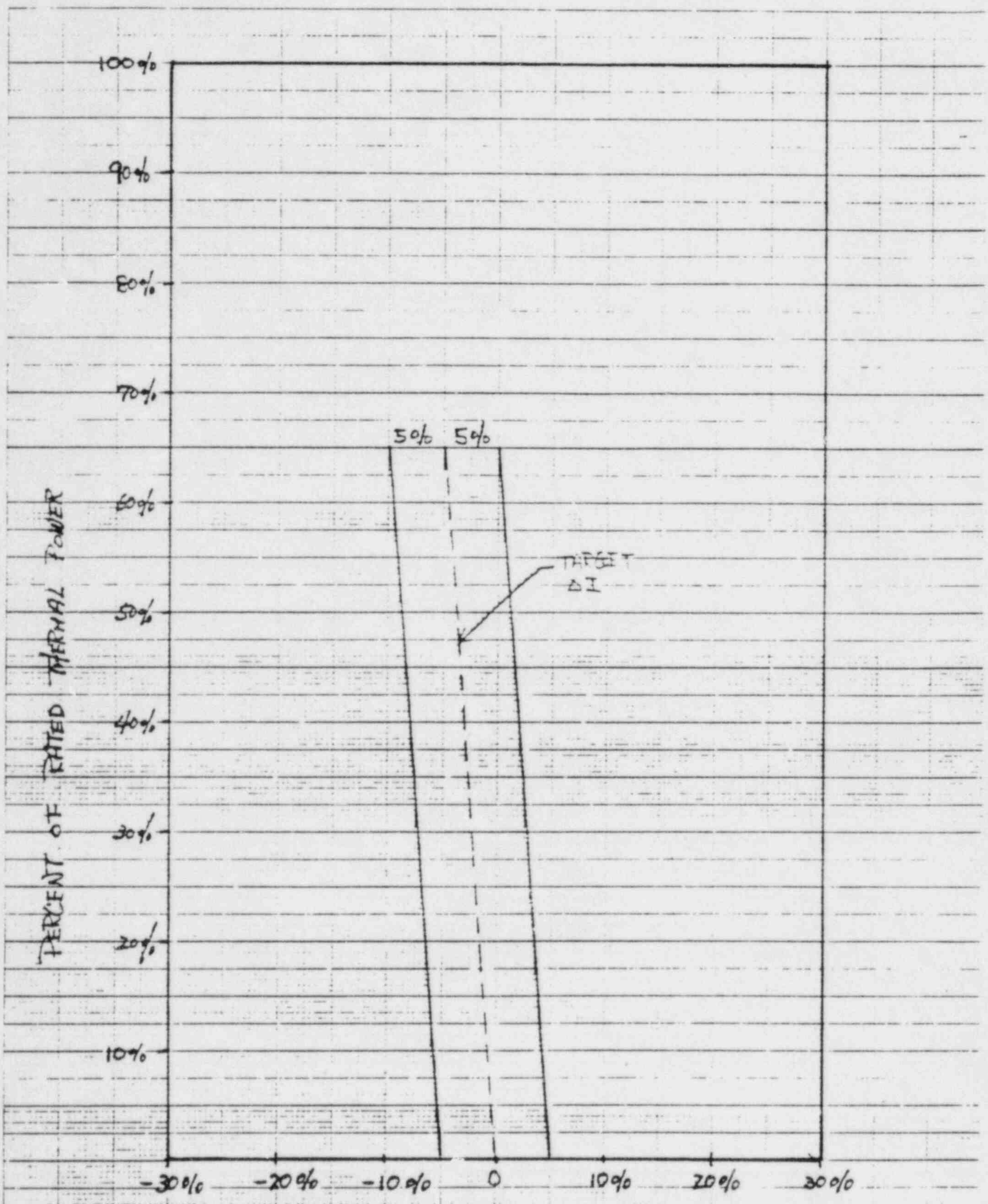


FIGURE B3/4 2-1a TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER



TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER FOR THREE LOOP OPERATION

Figure B3/4 2-1b

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.

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

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. ~~As noted on Figures 3.2-3 and 3.2-4, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value.~~ 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.

INSERT G

~~R_1 as calculated in 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$~~ less than or equal to 1.49. This value is 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. R_2, as defined, allows for the inclusion of a penalty for rod bow on DNBR only. Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "tradeoffs" in excess of R equal to 1.0 for the purpose of offsetting the rod bow DNBR penalty.

Fuel rod bowing reduces the value of DNB ratio. Sufficient credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% and 3% margin in the difference between the 1.3 DNBR safety limit and the minimum DNBR calculated for the Complete Loss of Flow event. The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 and W 8691 ~~Rev. 1 (partial rod bow test data).~~

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.

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The $F_{\Delta H}^{N}$ as calculated in section 3.2.3 are used in various accident analyses where $F_{\Delta H}^{N}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum value allowed ~~(measured plus uncertainty)~~. ment

as measured

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

1) Design limit DNBR Of 1.30 vs. 1.28
2) Grid Spacing (K_s) of 0.046 vs. 0.059
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059
4) DNBR Multiplier of 0.86 vs. 0.88
5) 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)

INSERT 6.1

~~When RCS flow rate and~~ $\cancel{F^N_{\Delta H}}$ ~~are measured, no additional allowances are necessary prior~~ to comparison with the limits ~~of Figures 3.2-3 and 3.2-4.~~ in the LCO. ~~Measurement errors of 3.5% for RCS total flow rate and 4% for~~ $\cancel{F^N_{\Delta H}}$ ~~have been allowed for in determination of the design DNBR value.~~

~~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 operation shown on Figure 3.2-3.~~

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.

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 two-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 percent for each percent of tilt in excess of 1.0.

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 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.

INSERT 6.1

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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}) 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 ~~Figures 3.2-3 and 3.2-4.~~ Measurement errors of ~~[2.1]%~~ 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 detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi ~~is included in Figure 3.2-3.~~ will be added if venturies are not verified clean every 18 months. Any fouling which might bias the RCS flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation ~~shown on Figure 3.2-3.~~ limits defined in Section 3.2-3.

~~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.~~

~~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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.~~

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-70

SPECIFICATION: 3/4.3.1 Reactor Trip System Instrumentation

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Modified the specification to incorporate three loop operation as designed for Unit 3.

2. Modified action time statements and surveillance intervals to incorporate WCAP-10271, Evaluation of Surveillance Frequencies and Out of Service Time for the Reactor Protection Instrumentation Systems.

PORC COMMENTS:

1. It was noted that cautionary notes would be incorporated into Table 4.3-1 that refer to the more stringent ESFAS requirements for analog signals that input into both RTS and ESFAS. This is a requirement of the NRC safety evaluation for WCAP-10271.

TS-70 (ADDENDUM)

1. Specification was modified to conform with the recommendations in Generic letter 85-09, Technical Specifications for Generic Letter 83-28, Item 4.3. The major changes include adding the reactor trip bypass breakres to the specifications and clearly defining the requirements to independently test the automatic shunt trip feature.

2. PORC reviewed the commitments necessary to take credit for WCAP-10271. The commitments are as follows:
   a. Unit 3 must confirm that a program exists which will address the identification of and required actions associated with plausible common cause problems as related to reactor protection system surveillance. ACP-QA-10.01, Plant Incident Reports, provides the mechanism to implement this. It was noted that a Unit 3 specific section will be added which requires the performance and documentation of a common cause failure analysis as part of the investigation of reactor protection system surveillance PIR's.

   b. Unit 3 will collect the "as found" and "as left" data for each channel over a one year period after quarterly testing has begun. Any necessary changes to setpoints and allowable values will be made after the data has been reviewed.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	11
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3 #
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	~~2~~ 4	0	1	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	11
7. Overtemperature ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	~~4~~ 3	~~1**~~ 2	~~3~~ 2	1, 2	~~9~~ 6#
8. Overpower ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	~~4~~ 3	~~1**~~ 2	~~3~~ 2	1, 2	~~9~~ 6#
9. Pressurizer Pressure--Low	4	2	3	1**	6# ****
10. Pressurizer Pressure--High	4	2	3	1, 2	6# ****
11. Pressurizer Water Level--High	3	2	2	1 **	6#

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop in each operating loop	2/loop in any operating loop	2/loop in each operating loop	1	6/#
b. Two Loops (Above P-7 and below P-8)	3/loop in each operating loops	2/loop in two operating loops	2/loop each operating loop	1	6/#
13. Steam Generator Water Level--Low-Low	4/stm. gen. in each operating stm gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6# ****
14. Low Shaft Speed--Reactor Coolant Pumps	~~4 1/pump~~ INSERT	~~2~~	~~3~~	~~1~~	~~6#~~
15. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1***	~~7#~~ 12#
b. Turbine Stop Valve Closure	4	4	4	1***	~~7#~~ 6#
16. Safety Injection Input from ESF	2	1	2	1, 2	10
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8

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FUNCT. UNIT	TOT # CHANNELS	CHANNELS TO TRIP	MIN CHANNELS OPERABLE	APP. MODES	ACTION
14. REACTOR COOLANT PUMP LOW SHAFT SPEED					
A. FOUR LOOPS OPERATING	4 - 1/PUMP IN EACH OPERATING LOOP	2	3	1**	6#
B. THREE LOOPS OPERATING	3 - 1/PUMP IN EACH OPERATING LOOP	2	2	1**	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1,2	8
~~f. Turbine Impulse Chamber Pressure, P-13~~	~~2~~	~~1~~	~~2~~	~~1~~	~~8~~
18. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3*, 4*, 5*	10 11
19. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3*, 4*, 5*	10 11
20 THREE LOOP OPERATION BYPASS CIRCUITRY	8 - 1 switch per loop in each train	2 Different loop switches in bypass	8	1, 2	1
21 REACTOR TRIP BYPASS BREAKER	2 2	1 1	2 2	1, 2 3*, 4*, 5*	10 11

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**** Comply with the provisions of Specification 3.3.2 for any portion of the channel required to be OPERABLE by Specification 3.3.2.

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

**~~The channel(s) associated with the trip functions derived from the out of service reactor coolant loop shall be placed in the tripped condition.~~ Above the P-7 (at power) setpoint

#The provisions of Specification 3.0.4 are not applicable.

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

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

*** Above the P-9 (Reactor trip/Turbine trip interlock) setpoint.

ACTION STATEMENTS

ACTION 1 - 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 be in HOT STANDBY within the next 6 hours.

ACTION 2 - 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 ~~1~~ 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

for 4 loop operation or 50% of RATED THERMAL POWER for 3 loop operation

for 4 loop operation or 60% of RATED THERMAL POWER for 3 loop operation

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - 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, suspend all operations involving positive reactivity changes and verify Valve~~s~~ 3CHS-V305 ~~and 3CHS-V304~~ are closed and secured in position within the next hour.

ACTION 6 - 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 ~~1~~ 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

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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 4 ~~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,

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

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	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ 4 seconds*
8. Overpower ΔT	≤ 4 seconds*
9. Pressurizer Pressure--Low	≤ 2 seconds
10. Pressurizer Pressure--High	≤ 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. (This provision is not applicable to CPs docketed after January 1, 1978. See Regulatory Guide 1.118, November 1977.)

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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)	$\leq$ 1 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1 second
13. Steam Generator Water Level--Low-Low	$\leq$ 2 seconds
14. Low Shaft Speed-Reactor Coolant Pumps	$\leq$ 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.
21. Reactor Trip Bypass Breakers	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.

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TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R (11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	~~M~~ Q (15)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	~~M~~ Q (15)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	~~M~~ Q (15)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	~~M~~ Q (15)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1), ~~M~~ Q (15)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S (14)	R(4, 5, ~~12~~ 14)	S/U(1), ~~M~~ Q(9)(15)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R(~~13~~ 12)	~~M~~ Q (15)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	~~M~~ Q (15)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	~~M~~ Q (15, 16)	N.A.	N.A.	1****
10. Pressurizer Pressure--High	S	R	~~M~~ Q (15, 16)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	~~M~~ Q (15)	N.A.	N.A.	1****
12. Reactor Coolant Flow--Low	S	R	~~M~~ Q (15)	N.A.	N.A.	1****

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	TRIP MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
20 Three Loop Operation Bypass Ciruitry	NA	NA	NA	R	NA	1,2
21 Reactor Trip Bypass Breaker	NA	NA	NA	M (18) R(19)	NA	1,2,3*,4*,5*

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**** Above P-7 (Reactor at Power) Setpoint

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint

* 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 ~~7~~ 92 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) With power greater than or equal to the Interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) ~~Monthly~~ 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. ~~Monthly~~ Surveillance shall include verification of the ~~Boron Dilution Alarm Setpoint~~ High Flux at Shutdown alarm setpoint of less than or equal to ~~(an increase of twice the count rate within a 10-minute period).~~ 5X background.

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) ~~At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.~~

~~(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves close and the centrifugal charging pumps suction valves from the RWST open within [30] seconds.~~

12 ~~(13)~~ CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

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

(14.) Surveillance is required for Post Accident Neutron Flux Monitoring Channel when they are used to meet the Minimum Number of Channel requirement of Table 3.3-2.6.c. ANALOG CHANNEL OPERATIONAL TEST is not required for Post Accident Neutron Flux Monitors.

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

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

(16.) Comply with the surveillance requirements of Specification 4.3.2.1 for any portion of the channel required to be OPERABLE by Specification 3.3.2.

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

(18) Local manual shunt trip prior to placing breaker in service. ~~(Or for plants that do not actuate the shunt trip attachment of the bypass breakers on a manual reactor trip): Remote manual undervoltage trip when breaker placed in service.~~

(19) Automatic undervoltage trip.

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-71

SPECIFICATION: 3/4.3.3.6 Accident Monitoring Instrumentation

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Modified the Table 3.3-10 for reactor coolant hot and cold leg wide range temperature indicators to requre that they be in operating loops. This is a clarification for three loop operation. There were no other changes noted between this and the draft specification submitted to the NRC.

PORC COMMENTS:

None.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT ~~(Illustrational Only)~~	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure a. Normal Range b. Extended Range	2 2	1 1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	~~2~~ 1/loop in 2 operating loops	1/loop in an operating loop
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	~~2~~ 1/loops in 2 operating loops	1/loops in an operating loop
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10 Neutron Flux	2	1
~~10. Boric Acid Tank Solution Level~~	~~2~~	~~1~~
11. Auxiliary Feedater Flow Rate	2/steam generator	1/steam generator
~~12. Reactor Coolant System Subcooling Margin Monitor~~	~~2~~	~~1~~
~~13. PORV Position Indicator~~	~~2/Valve~~	~~1/Valve~~
~~14. PORV Block Valve Position Indicator~~	~~2/Valve~~	~~1/Valve~~
~~15. Safety Valve Position Indicator~~	~~2/Valve~~	~~1/Valve~~
~~16. Containment Water Level (Narrow Range)~~	~~2~~	~~1~~
17. Containment Water Level (Wide Range)	2	1
18. In Core Thermocouples	4/core quadrant	2/core quadrant

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-72

SPECIFICATION: 3/4.4.1.1 Reactor Coolant Loops - Startup and Power Operation

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Deleted specification 3/4.4.1.1 and added specifications 3/4.4.1.1.1 and 3/4.4.1.1.2. This was necessary to incorporate three loop operation. It was noted that new specification 3/4.4.1.1.1 is identical to the draft specification 3/4.4.1.1 submitted to the NRC.

2. New specification 3/4.4.1.1.2 was added to incorporate the thermal power limit for three loop operation.

3. Modified the bases to reflect three loop operation.

PORC COMMENTS:

1. The references to the loop stops in the LCO 3.4.1.1.2 and surveillance 4.4.1.1.2.2 was deleted as recommended by PORC. These requirements are duplicated in specification 3/4.4.1.6.

SPECIFICATION: 3/4.4.1.6 Isolated Loop Startup

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATION:

1. Modified specification 3/4.4.1.6 to conform to the design and analysis assumptions for three loop operation.

2. Changed shutdown margin to conform with the boron dilution analysis assumption.

3. Added surveillance 4.4.1.6.3 to insure that the isolated loop stop valves are shut with power removed.

PORC COMMENTS:
None.

SPECIFICATION: 3/4.4.5 Steam Generators

FSAR REFERENCE: ------

SER REFERENCE: ------

Transmittal No.: TS-72 (continued)

W STS DEVIATION:

1. Modified specification to require that only steam generators in unisolated loops be OPERABLE.

PORC COMMENTS:

None.

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

Four Loops Operating

LIMITING CONDITION FOR OPERATION

3.4.1.1 ~~3.4.1.1~~ All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 ~~4.4.1.1~~ The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

Three Loops Operating

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 ~~3.4.1.1~~ ~~All~~ Three reactor coolant loops shall be in operation with THERMAL POWER restricted to 65% of RATED THERMAL POWER

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.2 ~~4.4.1.1~~ The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

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ISOLATED LOOP

LIMITING CONDITION FOR OPERATION

3.4.1.5 The boron concentration of ~~an~~ a filled isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops.

APPLICABILITY: MODES ~~1, 2, 3, 4, and~~ 5. and 6

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loop's stop valves. ~~either increase the boron concentration of the isolated loop to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours with the unisolated portion of the RCS borated to a SHUTDOWN MARGIN equivalent to at least 2% Δk/k at 200°F.~~

SURVEILLANCE REQUIREMENTS

4.4.1.5 The boron concentration of an isolated loop shall be determined to be greater than or equal to the boron concentration of the operating loops ~~at least once per 24 hours and~~ within 30 minutes prior to opening either the hot leg or cold leg stop valves of an isolated loop.

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ISOLATED LOOP STARTUP

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LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated until:

a. The isolated loop ~~has been operating on a recirculation flow of greater than or equal to ___ gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop~~ is within 20°F of the highest cold leg temperature of the operating loops, and

b. The reactor is subcritical by at least 2% Δk/k.

c. The plant is in Mode 5 or 6.

APPLICABILITY: All MODES.

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

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 2% Δk/k within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.3 The loop stops in the non-operating loop shall be verified to be closed, with power to the valve operator removed, at least once per 31 days when all three conditions of 3.4.1.6 are not satisfied.

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3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

unisolated

3.4.5 Each steam generator shall be OPERABLE.

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

ACTION:

unisolated

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed in Modes 1 and 2 to operate with ~~all~~ three or four reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. ~~In MODES 1 and 2~~ With ~~one~~ less than the required reactor coolant loops not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to ~~(275)~~ 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by ~~either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by~~ restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of ~~an~~ a filled isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. ~~Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.~~ Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from the cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity [illegible]

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-73

SPECIFICATION: 3/4.7.1.1 Safety Valves

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Modified specification and bases to reflect three loop operation. It was noted that the only difference from the draft specification submitted to the NRC was the incorporation of the power range flux high trip setpoint for three loop operation.

PORC COMMENTS:

None.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator of an unisolated reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	42

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR*	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	~~[52]~~ 64
2	~~[38]~~ 48
3	~~[25]~~ 32

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

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TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± 1%)*	ORIFICE SIZE
LOOP 1		
RV22A	~~1200 psia~~ 1185 PSIG	16.0 square inches
RV23A	~~1210 psia~~ 1195 PSIG	16.0 square inches
RV24A	~~1220 psia~~ 1205 PSIG	16.0 square inches
RV25A	~~1230 psia~~ 1215 PSIG	16.0 square inches
RV26A	~~1240 psia~~ 1225 PSIG	16.0 square inches
LOOP 2		
RV22B	~~1200 psia~~ 1185 PSIG	16.0 square inches
RV23B	~~1210 psia~~ 1195 PSIG	16.0 square inches
RV24B	~~1220 psia~~ 1205 PSIG	16.0 square inches
RV25B	~~1230 psia~~ 1215 PSIG	16.0 square inches
RV26B	~~1240 psia~~ 1225 PSIG	16.0 square inches
LOOP 3		
RV22C	~~1200 psia~~ 1185 PSIG	16.0 square inches
RV23C	~~1210 psia~~ 1195 PSIG	16.0 square inches
RV24C	~~1220 psia~~ 1205 PSIG	16.0 square inches
RV25C	~~1230 psia~~ 1215 PSIG	16.0 square inches
RV26C	~~1240 psia~~ 1225 PSIG	16.0 square inches
LOOP 4		
RV22D	~~1200 psia~~ 1185 PSIG	16.0 square inches
RV23D	~~1210 psia~~ 1195 PSIG	16.0 square inches
RV24D	~~1220 psia~~ 1205 PSIG	16.0 square inches
RV25D	~~1230 psia~~ 1215 PSIG	16.0 square inches
RV26D	~~1240 psia~~ 1225 PSIG	16.0 square inches

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (~~1320 psia~~ 1304 PSIG) of its design pressure of ~~1200 psia~~ 1185 PSIG during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~____ lbs/h which is ____ % of the total secondary steam flow of ____ lbs/h at 100% RATED THERMAL POWER.~~ equal to 105% of the maximum calculated steam generator mass flow at setpoint pressure plus accumulation. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times \cancel{(76)}\ 80$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

U = Maximum number of inoperable safety valves per operating steam line,

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SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for ~~[N]~~ 4 loop operation,

~~[76]~~ 80 = ~~Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 3 loop operation,~~ Power Range Neutron Flux - High Trip Setpoint for 3 loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

~~3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM~~

~~The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.~~

~~Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of [350] gpm at a pressure of [1133] psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of [700] gpm at a pressure of [1133] psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.~~

~~3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK~~

~~The OPERABILITY of the demineralized water storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.~~

~~3/4.7.1.4 SPECIFIC ACTIVITY~~

~~The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.~~

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-74

SPECIFICATION: 3/4.10.2 Group Height, Insertion, and Power Distribution Limits

FSAR REFERENCE: ------

SER REFERENCE: ------

W STS DEVIATIONS:

1. Deleted specification 3/4.10.2 and added specifications 3/4.10.2.1 and 3/4.10.2.2. This was necessary due to the changes made to 3/4.2.1 and 3/4.2.2. It was noted that the intent of the specifications have not been changed from the draft specification submitted to the NRC.

PORC COMMENTS:

None.

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3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

Four Loops Operating

LIMITING CONDITION FOR OPERATION

3.10.2.1 ~~3.10.2~~ The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.1~~ 3.2.1.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications ~~3.2.2~~ 3.2.2.1 and ~~3.2.3~~ 3.2.3.1 are maintained and determined at the frequencies specified in Specification ~~4.10.2.2~~ 4.10.2.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification ~~3.2.2~~ 3.2.2.1 or ~~3.2.3~~ 3.2.3.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.1~~ 3.2.1.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specification. ~~3.2.2~~ 3.2.2.1 and ~~3.2.3~~ 3.2.3.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1.1 ~~4.10.2.1~~ The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.1.2 ~~4.10.2.2~~ The Surveillance Requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications ~~4.2.2.2~~ 4.2.2.1.2 and ~~4.2.2.3~~ 4.2.2.1.3, and

b. Specification ~~4.2.3.2~~ 4.2.3.1.2.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

Three Loops Operating

LIMITING CONDITION FOR OPERATION

~~3.10.2~~ 3.10.2.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.1~~ 3.2.1.2, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to ~~85%~~ 53% of RATED THERMAL POWER, and

b. The limits of Specifications ~~3.2.2~~ 3.2.2.2 and ~~3.2.3~~ 3.2.3.2 are maintained and determined at the frequencies specified in Specification ~~4.10.2.2~~ 4.10.2.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification ~~3.2.2~~ 3.2.2.2 or ~~3.2.3~~ 3.2.3.2 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.2.1~~ 3.2.1.2, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications ~~3.2.2~~ 3.2.2.2 and ~~3.2.3~~ 3.2.3.2, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

~~4.10.2.1~~ 4.10.2.2.1 The THERMAL POWER shall be determined to be less than or equal to ~~85%~~ 53% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

~~4.10.2.2~~ 4.10.2.2.2 The Surveillance Requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications ~~4.2.2.2~~ 4.2.2.2.2 and ~~4.2.2.3~~ 4.2.2.3.2, and

b. Specification ~~4.2.3.2~~ 4.2.3.2.2.

TECHNICAL SPECIFICATION
SUMMARY SHEET

TRANSMITTAL NO. TS-75

SPECIFICATION: 3/4.6.6.1 Supplementary Leak Collection and Release System

FSAR REFERENCE: Table 1.8-1
Table 1.9-2
6.2.3
6.5

SER REFERENCE: 6.2.3
6.5.1

W STS DEVIATIONS:

1. Modified surveillances 4.6.6.1.b.1), 4.6.6.1.b.2) and 4.6.6.1.C. The change consisted of deleting the STS paragraphs and replacing them with specific surveillance requirements from Regulatory Guide 1.52 positions C.5.a, C.5.c, C.5.d and C.6.b. This is necessary since the FSAR is being changed use ANSI N510-1980 vice ANSI N510-1976. In the case of position C.6.b, the surveillance specifies the filter test conditions as slated in the FSAR.

PORC COMMENTS:

1. It was noted that the acceptance criteris for surveillances 4.6.6.1.b.4 and 4.6.6.1.c has been changed from 0.175% to 1%. This results from the absorber efficiency of 95% assumed in the FSAR Chapter 15 analysis. The SER must be updated to reflect this assumption.

SPECIFICATION: 3/4.7.7.1 Control Room Emergency Air Filtration System

FSAR REFERENCE: Table 1.8-1
Table 1.9-2
6.5
9.4.1

SER REFERENCE 6.4
6.5.1
9.4.1

W STS DEVIATIONS:

1. Modified surveillances 4.7.7.1.c.1), 4.7.7.1.c.2) and 4.7.7.1.d. The change and reason are the same as deviation 1. for specification 3/4.6.6.1.

PORC COMMENTS:

1. Changed acceptance criteria for surveillances 4.7.7.1.c.4) and 4.7.7.1.d. as noted in PORC Comment 1. of specification 3/4.6.6.1.

2. It was noted that the filter flow rate was changed from 1000 CFM ±10% to 1120 CFM ±20%. Based on the discussion for Regulatory Guide 1.52 in Table 1.8-1, the bounding flows for this system are 1225 CFM ±10% with clean filters and 1000 CFM ±10% in the dirty filter condition. This is a system balancing/adjustment criteria. Since for system operation any flow between 1225 CFM +10% and 1000 CFM -10% is acceptable, an acceptance criteria of 1120 CFM ±20% was chosen for the surveillance.

SPECIFICATION: 3/4.7.8 Auxiliary Building Filter System

FSAR REFERENCE: Table 1.8-1
Table 1.9-2
6.5
9.4.2

SER REFERENCE: 6.5.1
9.4.2

W STS DEVIATIONS:

1. Modified surveillances 4.7.8.b.1), 4.7.8.b.2) and 4.7.8.C. The change and reason are the same as deviation 1. for specification 3/4.6.6.1.

PORC COMMENTS:

1. Changes acceptance criteria for surveillances 4.7.8.b.4) and 4.7.8.c as noted in PORC Comment 1. of specification 3/4.6.6.1.

SPECIFICATION: 3/4.9.12 Fuel Building Exhaust Filter System

FSAR REFERENCE: Table 1.8-1
6.5
9.4.2

SER REFERENCE: 6.5.1
9.4.2

W STS DEVIATIONS:

1. Modified surveillances 4.9.12.b.1), 4.9.12.b.2) and 4.9.12.c. This change and reason are the same as deviation 1. for specification 3/4.6.6.1.

2. It was noted that in the December 1984 submittal of the draft technical specifications the LCO and Action were modified and a surveillance added to reflect the plant design. The system receives no automatic actuation signals; thus, the STS LCO is not appropriate. The new LCO is consistent with the intent of the STS Action a which is to have one train

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OPERABLE and in operation when moving fuel within or loads over the spent fuel. New Action a is consistent with the intent of STS Action b. The added surveillance insures LCO compliance.

PORC COMMENTS:

1. Changes acceptance criteria for surveillances 4.9.12.c.4) and 4.9.12.d as noted in PORC comment 1. of specification 3/4.6.6.1.

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CONTAINMENT SYSTEMS

3/4.6.~~8~~6 SECONDARY CONTAINMENT

~~SHIELD BUILDING AIR CLEANUP SYSTEM~~

SUPPLEMENTARY LEAK COLLECTION & RELEASE SYSTEM (SLCRS)

LIMITING CONDITION FOR OPERATION

3.6.~~8~~6.1 Two independent ~~Shield Building Air Cleanup Systems~~ SLCRS systems shall be OPERABLE. | C

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

ACTION:

With one ~~Shield Building Air Cleanup System~~ SLCRS inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. | C

SURVEILLANCE REQUIREMENTS

4.6.~~8~~6.1 Each ~~Shield Building Air Cleanup System~~ SLCRS shall be demonstrated OPERABLE: | C

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating; | A

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

INSERT 1

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ________ cfm ± 10%;~~

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and~~ | A

5 ~~3~~) Verifying a system flow rate of 9500 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975.~~ 1980 | C

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INSERT 2

~~c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;~~ F

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~[6]~~ 6.25 inches Water Gauge while operating the system at a flow rate of 9500 cfm ± 10%, C

2) Verifying that the system starts on a Safety Injection test signal, C

~~3) Verifying that the filter cooling bypass valves can be manually opened,~~ C

3 ~~4~~) Verifying that each system produces a negative pressure of greater than or equal to [0.25] inch Water Gauge in the annulus within [1] minute after a start signal, and C

4 ~~5~~) Verifying that the heaters dissipate 50 ± 5 kW when tested in accordance with ANSI N510-~~1975~~ 1980. C

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the system at a flow rate of 9500 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9500 cfm ± 10%. A

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

INSERT 1

4.6.6.1.b.1) Performing a visual examination of SLCRS in accordance with ANSI N510-1980.

4.6.6.1.b.2) Verifying the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 9500 CFM ± 10%.

4.6.6.1.b.3) Verifying the charcoal absorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9500 CFM ±10%.

4.6.6.1.b.4 Verifing within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrate a methyl iodide penetration of less than 1% when tested in accordance with ANSI N510-1980 at 80°C, 70% R.H. and a face velocity of 47 FPM.

INSERT 2

4.6.6.1.c After every 720 hours of charcoal absorber operation by verifing within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrate a methyl iodide penetration of less than 1% when tested in accordance with ANSI N510-1980 at 80°C, 70% R.H. and a face velocity of 47 FPM.

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3/4.6.~~8~~ 6 SECONDARY CONTAINMENT

3/4.6.~~8~~ 6.1 VENTILATION SYSTEM Supplementary Leak Collection & Release System

The OPERABILITY of the ~~Shield Building Ventilation Systems~~ ensures that containment ~~vessel~~ leakage occurring during LOCA conditions into the ~~annulus~~ Enclosure Building and contiguous areas will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. Cumulative operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. This requirement is necessary to meet the assumptions used in the safety analyses and limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during LOCA conditions. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing. & C A

~~3/4.6.8.2~~ CONTAINMENT INTEGRITY

Secondary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Secondary Containment Ventilation System, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions. C

3/4.6.8.3 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability. C

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3/4.7.7 CONTROL ROOM EMERGENCY ~~AIR CLEANUP~~ VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 Two independent Control Room Emergency Air ~~Cleanup~~ Filtration Systems shall be OPERABLE. | C

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Emergency Air ~~Cleanup~~ Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. | C

MODES 5 and 6:

a. With one Control Room Emergency Air ~~Cleanup~~ Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Air ~~Cleanup~~ Filtration System in the recirculation mode. | C | C

b. With both Control Room Emergency Air ~~Cleanup~~ Filtration Systems Filtration inoperable, or with the OPERABLE Control Room Emergency Air ~~Cleanup~~ System required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes. | C

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each Control Room Emergency Air ~~Cleanup~~ Filtration System shall be demonstrated OPERABLE: | C

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to ~~[80]~~ 95°F; | J

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating; | A

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revisions 2, March 1978, and the system flow rate is ______ cfm ± 10%;~~ A

INSERT 1

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and~~

5 ~~3~~) Verifying a system flow rate of 1120 cfm ± ~~10%~~ 20% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980. C

d. ~~After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;~~ A

INSERT 2

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~[6]~~ 6.75 inches Water Gauge while operating the system at a flow rate of 1120 cfm ± ~~10%~~ 20%; C C C

~~2) Verifying that on a Containment Phase "A" Isolation and High Smoke Density test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;~~ J

2 ~~3~~) Verifying that the system maintains the control room at a positive pressure of greater than or equal to ~~[~~1/8~~]~~ inch Water Gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas during system operation; J

3 ~~4~~) Verifying that the heaters dissipate 9.4 ± 1 kW when tested in accordance with ANSI N510-~~1975~~ 1980; and C

~~5) Verifying that on a High Chlorine/Toxic Gas test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks within [15] seconds.~~ J

Insert 1

4.7.7.1.c.1) Performing a visual examination of the Control Room Emergency Air Filtration System in accordance with ANSI N510-1980.

4.7.7.1.c.2) Verifying the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-198 for a DOP test aerosol while operating the system at 1120 CFM ± 20%.

4.7.7.1.c.3) Verifying the charcoal absorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-198 for a halogenated hydrocarbon refrigerant test gas while operating the system at 1120 CFM ± 20%

4.7.7.1.c.4) Verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrate a methyl iodide penetration of less than 1% when tested in

accordance with ANSI N510-1980 at 80°C, 70% R.H. and a face velocity of 47 FPM.

INSERT 2

4.7.7.1.d After every 720 hours of charcoal absorber operation by verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canister demonstrates a methyl iodide penetration of less than 1% when tested in accordance with ANSI N510-1980 at 80°C, 70% RH and a face velocity of 47 FPM.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510~~-1975~~ for a DOP test aerosol while operating the system at a ~~flow rate~~ of ~~1120~~ 1980 cfm ± ~~10%~~ 20%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]%~~ 0.05% in accordance with ANSI N510-~~1975~~ for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of ~~1120~~ 1980 cfm ± ~~10%~~ 20%.

A

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

A

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~~ULTIMATE HEAT SINK (Conti~~nued)

~~The limitations on minimum water level~~ and maximum temperature are based on ~~providing~~ a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 FLOOD PROTECTION [OPTIONAL]

The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation ____ Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to ~~safety-related equipment.~~

3/4.7.7 CONTROL ROOM EMERGENCY ~~AIR CLEANUP~~ VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency ~~Air Cleanup~~ Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent for the duration of the accident. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-~~1975~~ 1980 will be used ~~as a~~ procedural guide for surveillance testing. C A

~~3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM~~

~~The OPERABILITY of the ECCS Pump Room Exhaust Air Cleanup System ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.~~ C A

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 AUXILIARY BUILDING FILTER SYSTEM

3/4.7.8 ~~ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM~~

LIMITING CONDITION FOR OPERATION

AUXILIARY BUILDING FILTER SYSTEMS

3.7.8 Two independent ~~ECCS Pump Room Exhaust Air Cleanup Systems~~ shall be OPERABLE. | C

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

ACTION:

AUXILIARY BUILDING FILTER

With one ~~ECCS Pump Room Exhaust Air Cleanup~~ System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. | C

SURVEILLANCE REQUIREMENTS

AUXILIARY BUILDING FILTER

4.7.8 Each ~~ECCS Pump Room Exhaust Air~~ Cleanup System shall be demonstrated OPERABLE: | C

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating; | A

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

INSERT 1

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ______ cfm ± 10%;~~ | A

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and~~

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SURVEILLANCE REQUIREMENTS (Continued)

5 ~~6)~~ Verifying a system flow rate of 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975.~~ 1980 C

INSERT 2

~~c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;~~ A

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~[6]~~ 6.8 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10%, C C

2) Verifying that the system starts on a Safety Injection test signal,

~~3) Verifying that the system maintains the ECCS pump room at a negative pressure of greater than or equal to [1/8] inch Water Gauge relative to the outside atmosphere,~~ C

~~4) Verifying that the filter cooling bypass valves can be manually opened, and~~ C

3 ~~5)~~ Verifying that the heaters dissipate 180 ± 18 kW when tested in accordance with ANSI N510-~~1975.~~ 1980 C

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% ~~[*]~~% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the system at a flow rate of 30,000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% ~~[*]~~% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30,000 cfm ± 10%. A

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

INSERT 1

4.7.8.b.1) Performing a visual examination of the Auxiliary Building Filter System in accordance with ANSI N510-1980.

4.7.8.b.2) Verifying the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at 30,000 CFM ± 10%

4.7.8.b.3) Verifying the charcoal absorber bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-19[illegible] for a halogenated hydrocarbon refrigerant test gas while operating the system at 30,000 CFM ± 10%

4.7.8.b.4) Verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrates a methyl iodide penetration of less

than 1% when tested in accordance with ANSI N510-1980 at 80°C, 70% R.H. and a face velocity of 47 FPM.

INSERT 2

4.7.8.c After every 720 hours of charcoal absorber operation by verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrates a methyl iodide penetration of less than 1% when tested in accordance with ANSI N510-1980 at 80°C, 70% R.H. and a face velocity of 47 FPM.

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~~ULTIMATE HEAT SINK~~ (Continued)

~~The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.~~

3/4.7.6 FLOOD PROTECTION [OPTIONAL]

~~The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation ___ Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety-related equipment.~~

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the Control Room Emergency Air Cleanup System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ~~ANSI N510-1975 will be used as a procedural guide for surveillance testing.~~ C A

AUXILIARY BUILDING FILTER

3/4.7.8 ~~ECCS PUMP ROOM EXHAUST AIR CLEANUP~~ SYSTEM

The OPERABILITY of the ~~ECCS Pump Room Exhaust Air Cleanup~~ Auxiliary Building Filter System ensures that radioactive materials leaking from the ~~ECCS~~ equipment within the ~~pump room~~ following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing. C A

{charging pump, component cooling water pump and heat exchanger areas

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3/4.9.12 ~~FUEL STORAGE POOL AIR CLEANUP SYSTEM~~ FUEL BUILDING EXHAUST FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 ~~Two independent Fuel Storage Pool Air Cleanup Systems shall be OPERABLE.*~~ | C

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

INSERT A

a. With one Fuel Storage Pool Air Cleanup System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Storage Pool Air Cleanup System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers. | C

b. With no Fuel Storage Pool Air Cleanup System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Storage Pool Air Cleanup System is restored to OPERABLE status. | C | C

~~c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.~~ | C

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel ~~Storage Pool~~ BUILDING EXHAUST Air ~~Cleanup~~ FILTER Systems shall be demonstrated OPERABLE: | C

INSERT B

b ~~a.~~ At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating; | A

c ~~b.~~ At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

INSERT A

3.9.12 At least one fuel building exhaust filter system shall be OPERABLE and in operation whenever any evolution involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool is in progress.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With no fuel building exhaust filter system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until at least one fuel building exhaust filter system is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

INSERT B

a. The final building exhaust filter system should be verified to be operating within 2 hours prior to the initiation of and at least once per 12 hours during either fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool.

INSERT C

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ____ cfm ± 10%;~~ A

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and~~

5. ~~3)~~ Verifying a system flow rate of 20,700 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980. C

INSERT D

~~c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%.~~ A

e ~~d.~~ At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~[6]~~ 6.2 inches Water Gauge while operating the system at a flow rate of 20,700 cfm ± 10%, C

~~2) Verifying that on a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,~~ C

INSERT C

4.9.12.c.1) Performing a visual examination of the Fuel Building Exhaust Filter System in accordance with ANSI N510-1980.

4.9.12.c.2) Verifying the HEPA filter bank satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at 20,700 CFM ± 10%.

4.9.12.c.3) Verifying the charcoal absorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at 20,700 CFM ± 10%

4.9.12.c.4) Verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canister demonstrates a methyl iodide penetration of less than 1% when tested in accordance with ANSI N510-1980 at 80°C ~~and~~ 70% R.H. and a face velocity of 47 FPM.

INSERT D

4.9.12.d After every 720 hours of charcoal absorber operation by verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canister demonstrates a methyl iodide penetration of less than 1% when tested in accordance with ANSI N510-1980 at 80°C, 70% RH and a face velocity of 40 FPM.

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SURVEILLANCE REQUIREMENTS (Continued)

2 ~~3)~~ Verifying that the system maintains the spent fuel storage pool area at a negative pressure ~~of greater than or equal to [1/4] inch Water Gauge~~ relative to the outside atmosphere during system operation, | C | C

~~4) Verifying that the filter cooling bypass valves can be manually opened, and~~ | C

3 ~~5)~~ Verifying that the heaters dissipate 150 ± 15 kW when tested in accordance with ANSI N510-~~1975.~~ 1980 | C

f. ~~e.~~ After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[*]~~% in accordance with ANSI N510-~~1975~~ 1980 for a DOP test aerosol while operating the system at a flow ~~rate~~ of 20,700 cfm ± 10%. 0.05% A

g. ~~f.~~ After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than ~~[**]~~% 0.05% in accordance with ANSI N510-~~1975~~ 1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 20700 cfm ± 10%.

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% of less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).~~ A

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

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BASES

~~3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL~~

~~The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.~~ C

FUEL BUILDING EXHAUST FILTER

3/4.9.12 ~~STORAGE POOL VENTILATION~~ SYSTEM

The limitations on the Fuel Building Exhaust Filter ~~Storage Pool Ventilation~~ System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing. C A

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

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

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

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

   Where:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection ~~(Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)~~.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	~~15~~ 16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12
c. Containment Pressure--High-1	3	2	2	1, 2, 3	~~15*~~ 13*
d. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	~~20*~~ 17*
~~e. Differential Pressure Between Steam Lines--High~~				1, 2, 3**	
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		15*
~~Three Loops Operating~~	~~3/operating steam line~~	~~1***/steam line any operating steam line~~	~~2/operating steam line~~		~~16~~

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water) (Continued)					
2) Three Loop Plant					
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		15*
Two Loops Operating	3/operating steam line	2***/steam line twice in either operating steam line	2/operating steam line		16
f. Steam Line Pressure--Low				1, 2, 3**	
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~1 pressure loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 3 loops~~		~~15*~~
~~Three Loops Operating~~	~~1 pressure/operating loop~~	~~1*** pressure in any operating loop~~	~~1 pressure in in any 2 operating loops~~		~~16~~
	3/STEAM LINE IN EACH OPERATING LOOP	2/STEAM LINE IN ANY OPERATING LOOP	2/STEAM LINE IN EACH OPERATING LOOP		13*

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TABLE 3.3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water) (Continued)~~					
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~1 pressure/ loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 2 loops~~		~~15*~~
~~Two Loops Operating~~	~~1 pressure/ loop~~	~~1*** pressure in any operating loop~~	~~1 pressure any operating loop~~		~~16~~
2. Containment Spray (CDA)					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	~~19~~ 16
b. Automatic Actuation Logic and Actuation Relays	2	1	2 [	1, 2, 3, 4	~~14~~ 12
c. Containment Pressure--High-3	4	2	3	1, 2, 3	~~17~~ 14

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FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	~~18~~ 16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12
3) Safety Injection Actuation Logic	2	1	2	1, 2, 3, 4	
	~~See Item 1. above for all Safety Injection initiating functions and requirements.~~				
b. Phase "B" Isolation					
1) Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	~~15~~ 16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12
3) Containment Pressure--High-3	4	2	3	1, 2, 3	~~17~~ 14
~~c. Purge and Exhaust Isolation~~					
~~1) Manual Initiation~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3, 4~~	~~18~~
~~2) Automatic Actuation Logic and Actuation Relays~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3, 4~~	~~18~~
~~3) Safety Injection~~	See Item 1. above for all Safety Injection initiating ~~functions and requirements.~~				

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	~~24~~ 21
2) System	2	1	2	1, 2, 3	~~23~~ 20
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	~~22~~ 19
c. Containment Pressure--High-2	3	2	2	1, 2, 3	~~15*~~ 13*
~~d. Steam Flow in Two Steam Lines--High~~				~~1, 2, 3~~	
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~2/steam line~~	~~1/steam line any 2 steam lines~~	~~1/steam line~~		~~15*~~
~~Three Loops Operating~~	~~2/operating steam line~~	~~1***/any operating steam line~~	~~1/operating steam line~~		~~16~~
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~2/steam line~~	~~1/steam line any 2 steam lines~~	~~1/steam line~~		~~15*~~
~~Two Loops Operating~~	~~2/operating steam line~~	~~1***/any operating steam line~~	~~1/operating steam line~~		~~16~~

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

DELETE

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~4. Steam Line Isolation (Continued)~~					
~~Steam Flow in Two Steam Lines--High~~					
~~Coincident With:~~					
T_{avg}--~~Low-Low~~				1, 2, 3	
~~1) Four Loop Plant~~					
Four Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops		15*
Three Loops Operating	1 T_{avg}/operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any two operating loops		16
2) ~~Three Loop Plant~~					
Three Loops Operating	~~1 T_{avg}/loop~~	1 T_{avg} any ~~2 loops~~	1 T_{avg} any 2 loops		15*
Two Loops Operating	1 T_{avg}/operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any operating loop		16

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TABLE 3.3' (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)	3/STEAM LINE IN EACH OPERATING LOOP	2/STEAM LINE IN ANY OPERATING LOOP	3/STEAM LINE IN EACH OPERATING LOOP		
e. Steam Line Pressure--Low				1, 2, 3**	13*
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~1 pressure/ loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 3 loops~~		~~15*~~
~~Three Loops Operating~~	~~1 pressure/ operating loop~~	~~1*** pressure in any operating loop~~	~~1 pressure in any 2 operating loops~~		~~16~~
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~1 pressure/ loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 2 loops~~		~~15*~~
~~Two Loops Operating~~	~~1 pressure/ operating loop~~	~~1*** pressure in any operating loop~~	~~1 pressure any operating loop~~		~~16~~
f. Steam Line Pressure - Negative Rate--High					3****
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15*~~
~~Three Loops Operating~~	~~3/operating steam line~~	~~2/steam line in any operating steam line~~	~~2/steam line in each operating steam line~~		~~16~~
	3/STEAM LINE IN EACH OPERATING LOOP	2/STEAM LINE IN ANY OPERATING LOOP	2/STEAM LINE IN EACH OPERATING LOOP	3	13*

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MILLSTONE - UNIT 3

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~4. Steam Line Isolation (Continued)~~					
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15~~
~~Two Loops Operating~~	~~3/operating steam line~~	~~2/steam line in any operating steam line~~	~~2/steam line in each operating steam line~~		~~16~~
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	~~25~~ 22
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen. in each operating Loop	2/stm. gen. in any operating ~~stm. gen.~~ Loop	3/stm. gen. in each operating ~~stm. gen.~~ Loop	1, 2	~~20*~~ 17*
6. Auxiliary Feedwater					
INSERT A					
~~a. Manual Initiation~~	2	1	2	1, 2, 3	~~23~~ 20
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	~~22~~ 19
~~c. Stm. Gen. Water Level--Low-Low~~					
Auxiliary Feedwater Pumps					
1. Steam Generator Low-Low Level					
~~1)~~ Start Motor-Driven Pumps	4/stm. gen. in each operating loop	2/stm. gen. in any operating stm. gen. Loop	3/stm. gen. in each operating ~~stm. gen.~~ Loop	1, 2, 3	~~20*~~ 17*

INSERT B

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
INSERT A					
5. TURBINE TRIP AND FEEDWATER ISOLATION					
c. SAFETY INJECTION ACTUATION LOGIC	2	1	2	1 & 2	19
d. Tavg Low Coincident with P-4					
Four Loops Operating	1 Tavg / Loop	1 Tavg in any 2 loops	1 Tavg in any 3 loops	1 & 2	17
Three Loops Operating	1 Tavg / Operating Loop	1 Tavg in any 2 Operating Loops	1 Tavg in any 2 Operating Loops	1 & 2	15

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

INSERT B

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (cont)					
• 2) Safety Injection Actuation (SIS) Logic	2	1	2	1, 2 & 3	19
• 3) Containment Depressurization Actuation (CDA) Logic	2	1	2	1, 2 & 3	19
• 4) Loss of Power (LOP) Logic	2	1	2	1, 2 & 3	19

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
~~c.~~ Start Turbine-Driven Pump STEAM GENERATOR LOW-LOW LEVEL	4/stm. gen. in each operating loop	2/stm. gen. in any 2 operating ~~stm. gen.~~ loop	3/stm. gen. in each operating ~~stm. gen.~~ loop	1, 2, 3	~~20*~~ 17*
~~d. Undervoltage RCP Start Turbine-Driven Pump~~	~~4-1/bus~~	~~2~~	~~3~~	~~1, 2~~	~~20*~~
~~d e. Safety Injection Start Motor-Driven Pumps~~	~~See Item 1. above for all Safety Injection initiating functions and requirements.~~				
~~e f. Loss of Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3~~	~~19~~
~~f g. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps and Turbine-Driven Pump~~	~~2/pump~~	~~1/pump~~	~~1/pump~~	~~1, 2~~	~~19~~
~~g h. Suction Transfer on Low Pressure~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~19~~

Control Building Isolation

~~7. Automatic Switchover to Containment Sump~~

INSERT C

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
c ~~a~~. Automatic Actuation Logic and Actuation Relays	2	1	2	ALL ~~1, 2, 3, 4~~	~~14~~
a. Manual Actuation	2	1	2	ALL	~~19~~
b. Manual Safety Injection [illegible]	2	1	2	1, 2, 3, 4	~~19~~

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

INSERT C

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. CONTROL BUILDING ISOLATION					
a. MANUAL ACTUATION	2	1	2	ALL	19
b. MANUAL SAFETY INJECTION ACTUATION	2	1	2	1, 2, 3 & 4	19
c. AUTOMATIC ACTUATION LOGIC AND ACTUATION RELAYS	2	1	2	ALL	12
d. CONTAINMENT PRESSURE HIGH-1	3	2	2	1, 2 & 3	13
e CONTROL BUILDING INLET VENTILATION RADIATION	1 [illegible]	1	1 [illegible]	ALL	15
f. OUTSIDE CHLORINE HIGH	[illegible]	1	1 [illegible]	ALL	15

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~7. Automatic Switchover to Containment Sump (Continued)~~					
~~b. RWST Level--Low Low~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3, 4~~	~~17~~
~~Coincident With:~~					
~~Containment Sump Level--High~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3, 4~~	~~17~~
~~and~~					
~~Safety Injection~~	~~See Item 1. above for all Safety Injection initiating functions and requirements.~~				
8. Loss of Power					
a. ~~4 kV~~ 4.16 kV Emergency Bus Under-voltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20*
b. ~~4 kV~~ 4.16 kV Emergency Bus Undervoltage-Grid Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20*
. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	~~21~~ 18
~~b. Low-Low T_{avg}, P-12~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~21~~
b. Low-Low Tave, P-12	1 Tave / LOOP	1 Tave / LOOP IN ANY 2 OPERATING LOOPS	1 Tave / LOOP IN ANY 2 OPERATING LOOPS	1, 2 & 3	18
c. Reactor Trip, P-4	2	2	2	1, 2, 3	~~23~~ 20
d. Steam Generator Water Level -- High-High, (P-14)	~~3~~ 4/stm. gen. in each operating Loop	2/stm. gen. in any operating ~~stm. gen.~~ Loop	~~2~~ 3/stm. gen. in each operating ~~stm. gen.~~ Loop	1, 2, 3	~~21~~ 18

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Emergency Generator Load Sequencer	2	1	2	1, 2, 3 & 4	12

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The Safety Injection Logic for this

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*The provisions of Specification 3.0.4 are not applicable.

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

**Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.

***The channel(s) associated with the protective functions derived from the out of service reactor coolant loop shall be placed in the tripped mode.

****Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION ~~14~~ 12 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION ~~15~~ 13 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

~~ACTION 16 - 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 and in at least HOT SHUTDOWN within the following 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.~~

ACTION ~~17~~ 14 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION ~~18~~ 15 - With less than the Minimum Channels OPERABLE requirement, ~~operation may continue provided the containment purge supply and exhaust valves are maintained closed.~~ within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION ~~18~~ 16 - 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 be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION ~~20~~ 17 - 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 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION ~~21~~ 18 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION ~~22~~ 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION ~~23~~ 20 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION ~~24~~ 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification [3.7.1.5].

ACTION ~~25~~ 22 - 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 4.3.2.1 provided the other channel is OPERABLE.

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TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection ~~(Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)~~					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1	3.3 ~~[3.0]~~	1.01 ~~[0.71]~~	1.75 ~~[1.5]~~	17.7 psia ≤ ~~[3.6] psig~~	18.5 psia ≤ ~~[3.86] psig~~
~~d. Pressurizer Pressure--Low~~	16.5 ~~[13.1]~~	13.67 ~~[10.71]~~	3.3 ~~[1.5]~~	1892 psia ≥ ~~[1850] psig~~	1885 psia ≥ ~~[1839] psig~~
~~e. Differential Pressure Between Steam Lines--High~~	~~[3.0]~~	~~[0.87]~~	~~[1.5/1.5]~~	≤ ~~[97] psi~~	≤ ~~[106] psi~~
f. Steam Line Pressure--Low	17.7 ~~[20.0]~~	15.31 ~~[10.71]~~	2.2 ~~[1.5]~~	659 psig ≥ ~~[675] psig~~	645 psig* ≥ ~~[635] psig*~~
2. Containment Spray					
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-3	3.3 ~~[3.0]~~	1.01 ~~[0.71]~~	1.75 ~~[1.5]~~	22.7 psia ≤ ~~[12.05] psig~~	23.5 psia ≤ ~~[12.31] psig~~

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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					
a. Phase "A" 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) Safety Injection Actuation Logic	NA ~~See Item 1. above for all~~ Safety Injection Trip Setpoints ~~and Allowable Values.~~	NA	NA	NA	NA
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 ~~[3.0]~~	1.01 ~~[0.71]~~	1.75 [1.5]	22.7 psia ≤ ~~[12.05] psig~~	23.5 psia ≤ ~~[12.31] psig~~
~~c. Purge and Exhaust 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) Safety~~ Injection	See Item 1. ~~above for all~~ Safety Injection Trip Setpoints ~~and Allowable Values.~~				

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ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
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	~~[3.0]~~ 3.3	~~[0.71]~~ 1.01	~~[1.5]~~ 1.75	~~[≤6.35] psig~~ ≤17.7 psia	~~[≤6.61] psig~~ ≤18.5 psia
~~d. Steam Flow in Two Steam Lines--High, Coincident with~~	~~[20.0]~~	~~[13.16]~~	~~[1.5/1.5]~~	~~≤ A function defined as follows: A ΔP corresponding to 40% of full steam flow between 0% and 20% load and then a ΔP increasing linearly to 110% of full load.~~	~~≤ A function defined as follows: A ΔP corresponding to 44% of full steam flow between 0% and load and then a ΔP increasing linearly to a ΔP corresponding to 114.0% of full steam flow at full load.~~
~~T_{avg}--Low-Low~~	~~[4.0]~~	~~[1.12]~~	~~[1.2]~~	~~≤[553]°F~~	~~≤[550.6]°F~~
d ~~f.~~ Steam Line Pressure--Low	~~[20.0]~~ 17.7	~~[10.71]~~ 15.31	~~[1.5]~~ 2.2	≤~~[675]~~ 659 psig	≤~~[635]~~ 645 psig*
e ~~f.~~ Steam Line Pressure - Negative Rate--High	~~[8.0]~~ 5.0	~~[0.5]~~ 0.5	~~[0]~~ 0	~~≤[110] psi/s~~ ≤-100 psi/s	~~≤ [121.6] psi/s**~~ ≤122.7 psi/s

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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
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 ~~[5.0]~~	2.39 [2.18]	1.75 [1.5]	≤82.0% <~~[82.4]%~~ of narrow range instrument span.	≤82.7% <~~[84.2]%~~ of narrow range instrument span.
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	17.2 ~~[30.0]~~	12.59 ~~[27.18]~~	1.75 ~~[1.5]~~	22.2% > ~~[32.2]%~~ of narrow range instrument span.	21.2% > ~~[30.4]%~~ of narrow range instrument span.
~~d. Undervoltage - RCP~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~< [70]% RCP bus voltage.~~	~~< [69]% RCP bus voltage.~~
d Safety Injection (SIS) Actuation Logic	NA	NA	NA	NA	NA
e Containment Depressurization Actuation Logic	NA	NA	NA	NA	NA
f Loss of Power (LOP) Logic	NA	NA	NA	NA	NA

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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)					
~~e. Safety Injection~~	~~See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.~~				
~~f. Loss-of-Offsite Power~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≥ [4800]V~~	~~≥ [4692]V~~
g. ~~Trip of All Main Feedwater Pumps~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	N.A.
h. ~~Suction Transfer on Low Pressure~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [442] ft~~	~~≤ [441] ft~~
~~7. Automatic Switchover to Containment Sump~~ Containment 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. ~~a.~~ ~~Automatic Actuation Logic and Actuation Relays~~	~~N.A.~~	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure - High-1	[ ]	[ ]	[ ]	[ ]	[ ]
~~b. RWST Level - Low-Low Coincident With Containment Sump Level~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [18]%~~ ~~≤ [30] in. above [680] ft~~	~~≤ [15]%~~ ~~≤ [32.5] in. above [680] ft~~
~~and Safety Injection~~	~~See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.~~				
8. Loss of Power					
a. 4.16 kV Emergency ~~4 kV~~ Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	N.A.	≤ [5760] volts with a ≤ [0.25] second time delay.	≤ [5652] volts with a ≤ [0.275] second time delay.
b. 4.16 kV Emergency ~~4 kV~~ Bus Undervoltage (Grid Degraded Voltage)	N.A.	N.A.	N.A.	≤ [6576] volts with a ≤ [3.3] second time delay.	≤ [6511] volts with a ≤ [3.3] second time delay.

INSERT D

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TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

INSERT D

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Control Building Isolation					
a. Manual Actuation	NA	NA	NA	NA	NA
b. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA
c. Manual Safety Injection Actuation	NA	NA	NA	NA	NA
d. Containment Pressure High-1					
e. Control Building Inlet Ventilation Radiation	***	***	***	***	***
f. Outside Chlorine High	****	****	****	****	****

See S&W

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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
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ ~~[1985] psig~~ 2015 psia	≤ ~~[1996] psig~~ 2025 psia
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ [553]°F	~~≥ [550.6]°F &~~ ≥ 551 and ≤ 555
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.~~				

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TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Emergency Generator Load Sequencer	NA	NA	NA	NA	NA

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq$ [50] seconds and $\tau_2 \geq$ [5] seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is less than or equal to [50] seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

*** For value see specification 3/4.3.3.1

**** For value see specification 3/4.3.3.7

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
~~e. Purge and Exhaust Isolation~~	~~N.A.~~
f. Steam Line Isolation	N.A.
~~g. Feedwater Isolation~~	~~N.A.~~
~~h. Auxiliary Feedwater~~	~~N.A.~~
~~i. Essential Service Water~~	~~N.A.~~
~~j. Containment Cooling Fans~~	~~N.A.~~
k. Control Room Isolation	N.A.
~~l. Reactor Trip~~	~~N.A.~~
~~m. Start Diesel Generator~~	~~N.A.~~
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	≤ [27](1)/[12](5) 5
1) Reactor Trip	≤ [2]
2) Feedwater Isolation	≤ ~~[7](3)~~ 6.8(3)
3) Phase "A" Isolation	≤ ~~[17](2)/[27](1)~~ 2/12 5
~~4) Purge and Exhaust Isolation~~	~~≤ [25](1)/[10](2)~~
5) Auxiliary Feedwater motor driven pump	≤ ~~[60]~~ 43.5
6) ~~Essential~~ Service Water	≤ [32](1)/[47](2)
~~7) Containment Cooling Fans~~	~~≤ [55](1)/[40](2)~~
8) Control ~~Room~~ Building Isolation	~~N.A.~~ 5.7
~~9) Start Diesel Generator~~	~~≤ [10]~~

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	5
a. Safety Injection (ECCS)	≤ [27](1)/[12](5)
1) Reactor Trip	≤ [2]
2) Feedwater Isolation	≤ ~~[7](3)~~ 6.8(3)
3) Phase "A" Isolation	≤ ~~[17](2)/[27](1)~~ 2(2)/12(1) + Note 5
~~4) Purge and Exhaust Isolation~~	~~≤ [25](1)/[10](2)~~
5) Auxiliary Feedwater	≤ ~~[60]~~ 43.5(5)/3.5(5)
6) ~~Essential~~ Service Water	≤ [47](1)/[32](2)
~~7) Containment Cooling Fans~~	~~≤ [55](1)/[40](2)~~
~~8) Control Room Isolation~~	~~N.A.~~
~~9) Start Diesel Generators~~	~~≤ [10]~~
~~4. Differential Pressure Between Steam Lines--High~~	
~~a. Safety Injection (ECCS)~~	~~≤ [22](4)/[12](5)~~
~~1) Reactor Trip~~	~~≤ [2]~~
~~2) Feedwater Isolation~~	~~≤ [7](3)~~
~~3) Phase "A" Isolation~~	~~≤ [17](2)/[27](1)~~
~~4) Purge and Exhaust Isolation~~	~~≤ [25](1)/[10](2)~~
~~5) Auxiliary Feedwater~~	~~≤ [60]~~
~~6) Essential Service Water~~	~~≤ [32](2)/[47](1)~~
~~7) Containment Cooling Fans~~	~~≤ [55](1)/[40](2)~~
~~8) Control Room Isolation~~	~~N.A.~~
~~9) Start Diesel Generators~~	~~≤ [10]~~
5. Steam Line Pressure--Low	5 4
a. Safety Injection (ECCS)	≤ [12](5)/[22](4)
1) Reactor Trip	≤ [2]
2) Feedwater Isolation	≤ ~~[7](3)~~ 6.8(3)
3) Phase "A" Isolation	≤ ~~[17](2)/[27](1)~~ 2(2)/12(1) + Note 6
~~4) Purge and Exhaust Isolation~~	~~≤ [25](1)/[10](2)~~
5) Auxiliary Feedwater	≤ ~~[60]~~ 43.5(1)
6) ~~Essential~~ Service Water	≤ [32](2)/[47](1)
~~7) Containment Cooling Fans~~	~~≤ [55](1)/[40](2)~~

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
Containment - High-3 (cont)	
Auxiliary Feedwater - Motor Driven Pumps	43.5
Essential Service Water	

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

ITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
Steam Line Pressure--Low (Continued)	
~~8) Control Room Isolation~~	~~N.A.~~
~~9) Start Diesel Generators~~	~~≤ [10]~~
b. Steam Line Isolation	≤ ~~[9]~~(3) 7
Containment Pressure--High-3	
a. Containment Spray	≤ ~~[45]~~ 3 (2)/~~[57]~~ 18 (1)
b. Phase "B" Isolation	≤ ~~[65]~~ 2 (1)/~~[75]~~ 12 (2)
INSERT	
Containment Pressure--High-2	
Steam Line Isolation	≤ ~~[9]~~ 17 (3)
~~Steam Flow in Two Steam Lines--High Coincident with T_{avg}--LowLow~~	
~~Steam Line Isolation~~	~~≤ [9](3)~~
Steam Line Pressure - Negative Rate--High	
Steam Line Isolation	≤ ~~[9]~~(3) 7
Steam Generator Water Level--High-High	
a. Turbine Trip	≤ [2.5]
b. Feedwater Isolation	≤ [7](3)
Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ [60]
b. Turbine-Driven Auxiliary Feedwater Pump	≤ [60]
~~Undervoltage RCP~~	
~~Turbine-Driven Auxiliary Feedwater Pump~~	~~≤ [60]~~
~~Loss of Offsite Power~~	
~~Turbine-Driven Auxiliary Feedwater Pump~~	~~≤ [60]~~
~~Trip of All Main Feedwater Pumps~~	
~~All Auxiliary Feedwater Pumps~~	~~N.A.~~

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
~~14 15. Suction Transfer on Low Pressure~~	
~~Auxiliary Feedwater (Suction Supply Automatic Realignment)~~	~~≤ [13]~~
~~16. RWST Level--Low-Low~~	
~~a. Automatic Switchover to Containment Sump~~	~~N.A.~~
~~b. Coincident with Containment Sump Level--High and Safety Injection (Automatic Switchover to Containment Sump)~~	~~≤[250](2)/[265](1)~~
5 ~~17~~. Loss of Power	
a. 4 kV Bus Undervoltage (Loss of Voltage)	≤ [10]
b. 4 kV Emergency Bus Undervoltage (Grid Degraded Voltage)	≤ [10]

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
Outside Chlorine High Control Building Isolation	4.0
Control Building Inlet Radiation Control Building Isolation	3.7
Low Tavg with Reactor Trip (P-4) Feedwater Isolation	6.8

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air-operated valves.

(4) Diesel generator starting and sequence loading delay included. RHR pumps not included.

(5) Diesel generator starting and sequence loading delays not included. RHR pumps not included.

(6) Time Required to close valves as indicated in Table 3.6-1

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection ~~(Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)~~								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~e. Differential Pressure Between Steam Lines-High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
f. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
C. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	NA	NA	NA	3R	M(1)	M(1)	Q	1,2,3,4
	~~See Item 1. above for all Safety Injection Surveillance Requirements~~.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~c. Purge and Exhaust Isolation~~								
~~1) Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
~~2) Automatic Actuation Logic and Actuation Relays~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2, 3, 4~~
~~3) Safety Injection~~	~~See Item 1. above for all Safety Injection Surveillance Requirements.~~							

TABLE 4.3-2 (Continued)

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ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation 1. System Level	NA	NA	NA	R	NA	NA	NA	1, 2, 3
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
c. Individual Line b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Flow in Two Steam Lines-High Coincident With~~	~~S~~	~~R~~	M	N.A.	~~N.A~~	N.A.	~~N.A~~	~~1, 2, 3~~
~~T_{avg}-Low-Low~~	~~S~~	~~R~~	M	N.A.	~~N.A.~~	N.A.	~~N.A~~	~~1, 2, 3~~
e. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
INSERT								
~~a. Automatic Actuation Logic and Actuation Relays~~	~~N.A.~~	N.A.	~~N.A.~~	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2~~
~~b. Steam Generator Water Level-High-High~~	S	R	M	N.A.	~~N.A.~~	N.A.	~~N.A.~~	~~1, 2~~
6. Auxiliary Feedwater								
~~a. Manual Initiation~~	N.A.	N.A.	N.A.	R	N.A.	N.A.	~~N.A.~~	~~1, 2, 3~~
~~b. Automatic Actuation and Actuation Relays~~	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	~~Q~~	~~1, 2, 3~~

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSERT

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. TURBINE TRIP AND FEEDWATER ISOLATION								
c. SAFETY INJECTION ACTUATION LOGIC	NA	NA	NA	R	NA	NA	NA	1, 2
d. TAVG LOW COINCIDENT WITH P-4	NA	R	Q	NA	NA	NA	NA	1,2
6. AUXILIARY FEEDWATER								
a. MOTOR DRIVEN AUXILIARY FEEDWATER PUMP								
1) STEAM GENERATOR WATER LEVEL LOW-LOW	S	R	Q	NA	NA	NA	NA	1,2&3
2) SAFETY INJECTION ACTUATION (SIS) LOGIC	NA	NA	NA	R	NA	NA	NA	1,2&3
3) CONTAINMENT DEPRESSURIZATION ACTUATION (CDA) LOGIC	NA	NA	NA	R	NA	NA	NA	1,2&3

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Feedwater Isolation and Turbine Trip								
a. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	M(1)	M(1)	Q	1 & 2
b. Safety Injection Actuation Logic	NA	NA	NA	R	NA	NA	NA	1 & 2
c. Steam Generator High-High Level	S	R	Q	NA	NA	NA	NA	1 & 2
d. T_{avg} Low Coincident with P-4	NA	R	Q	NA	NA	NA	NA	1 & 2

TABLE 3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater								
2. Motor Driven Auxiliary Feedwater Pumps								
1) Steam Generator Low-Low Level	S	R	Q	NA	NA	NA	NA	1, 2 & 3
2) Safety Injection Actuation (SIS) Logic	NA	NA	NA	R	NA	NA	NA	1, 2 & 3
3) Containment Depressurization Actuation (CDA) Logic	NA	NA	NA	R	NA	NA	NA	1, 2 & 3
4) Loss of Power (LOP) Logic	NA	NA	NA	R	NA	NA	NA	1, 2 & 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~6. Auxiliary Feedwater (Continued)~~								
~~c. Steam Generator Water Level-Low-Low~~	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
~~d. Undervoltage - RCP~~	~~N.A.~~	~~R~~	~~N.A.~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2~~
d ~~e~~. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e ~~f~~. ~~Loss-of-Offsite Power~~	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3
f ~~g~~. ~~Trip of All Main Feedwater Pumps~~	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
g ~~h~~. ~~Suction Transfer on Low Pressure~~	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~Control Building Isolation~~								
~~7. Automatic Switchover to Containment Sump~~								
a. Manual Actuation	N.A	N.A.	N.A.	R	N.A.	N.A.	~~N.A.~~	~~ALL~~
b. Manual Safety Injection Actuation	N.A	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
c ~~a~~. ~~Automatic Actuation Logic and Actuation Relays~~	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	~~Q~~	~~1, 2, 3, 4~~
~~b. RWST Level-Low-Low Coincident With~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A~~	~~1, 2, 3, 4~~
~~Containment Sump Level-High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
~~and~~								
~~Safety Injection~~	~~See Item 1. above for all Safety Injection Surveillance Requirements.~~							
~~i. Containment Pressure--High~~	S	R	M	N.A.	N.A.	N.A.	N.A.	2, 3

TABLE 4 2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Control Building Isolation								
a. Manual Actuation	NA	NA	NA	R	NA	NA	NA	All
b. Manual Safety Injection Actuation	NA	NA	NA	R	NA	NA	NA	1, 2, 3 & 4
c. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	M(1)	M(1)	Q	All
d. Containment High Pressure (Hi-1)	S	R	Q	NA	NA	NA	NA	1, 2 & 3
e. Control Building Inlet Ventilation Radiation	#	#	#	NA	NA	NA	NA	#
f. Outside Chlorine High	S	R	M	NA	NA	NA	NA	All

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power								
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	R.	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level, P-14	S	R	M	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATION

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

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
10 Emergency Generator Load Sequencer	NA	NA	NA	NA	Q(1,2)	NA	NA	1,2,3&4

2. This surveillance may be performed continuously by the Emergency Generator Load Sequencer Autotest System as long as the ESFS Autotest System is demonstrated OPERABLE by the performance of an ACTUATION LOGIC TEST at least once per 92 days.