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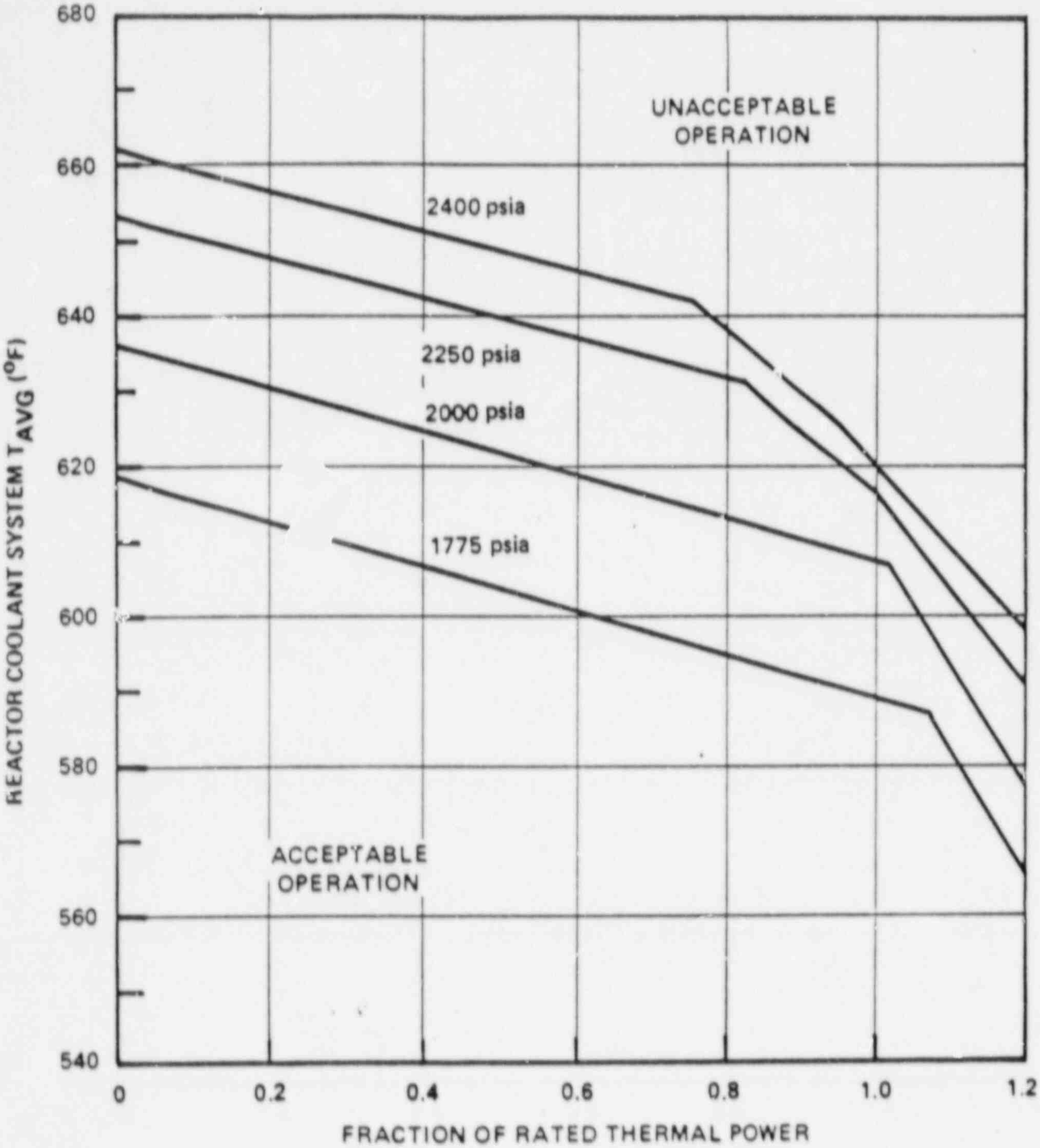

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

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 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 LOOP2 LOOP3 LOOP4 FI-0414 FI-0424 FI-0434 FI-0444 FI-0415 FI-0425 FI-0435 FI-0445 FI-0416 FI-0426 FI-0436 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)	18.5	17.18	1.67	≥18.5%*** of narrow range instrument span	≥17.8% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. Channel Calibration shall ensure that these time constants are adjusted to these values.

***Until resolution of the Veritrak transmitter uncertainty issue this setpoint will be set at ≥22.5% (Unit 1) and ≥23.5% (Unit 2) of narrow range instrument span.

≥

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

APPLICABILITY

LIMITING CONDITION FOR OPERATION (Continued)

3.0.5 Unless specifically noted, all the information provided in the Limiting Condition for Operation including the associated ACTION requirements shall apply to each unit individually. In those cases where a specification makes reference to systems or components which are shared by both units, the affected systems or components will be clearly identified in parentheses or footnotes declaring the reference to be "common." Whenever the Limiting Condition for Operation refers to systems or components which are common, the ACTION requirements will apply to both units simultaneously. (This will be indicated in the ACTION section.) Whenever certain portions of a specification refer to systems, components, operating parameters, setpoints, etc., which are different for each unit, this will be identified in parentheses or footnotes or in the APPLICABILITY section ~~as well.~~ as appropriate

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified.

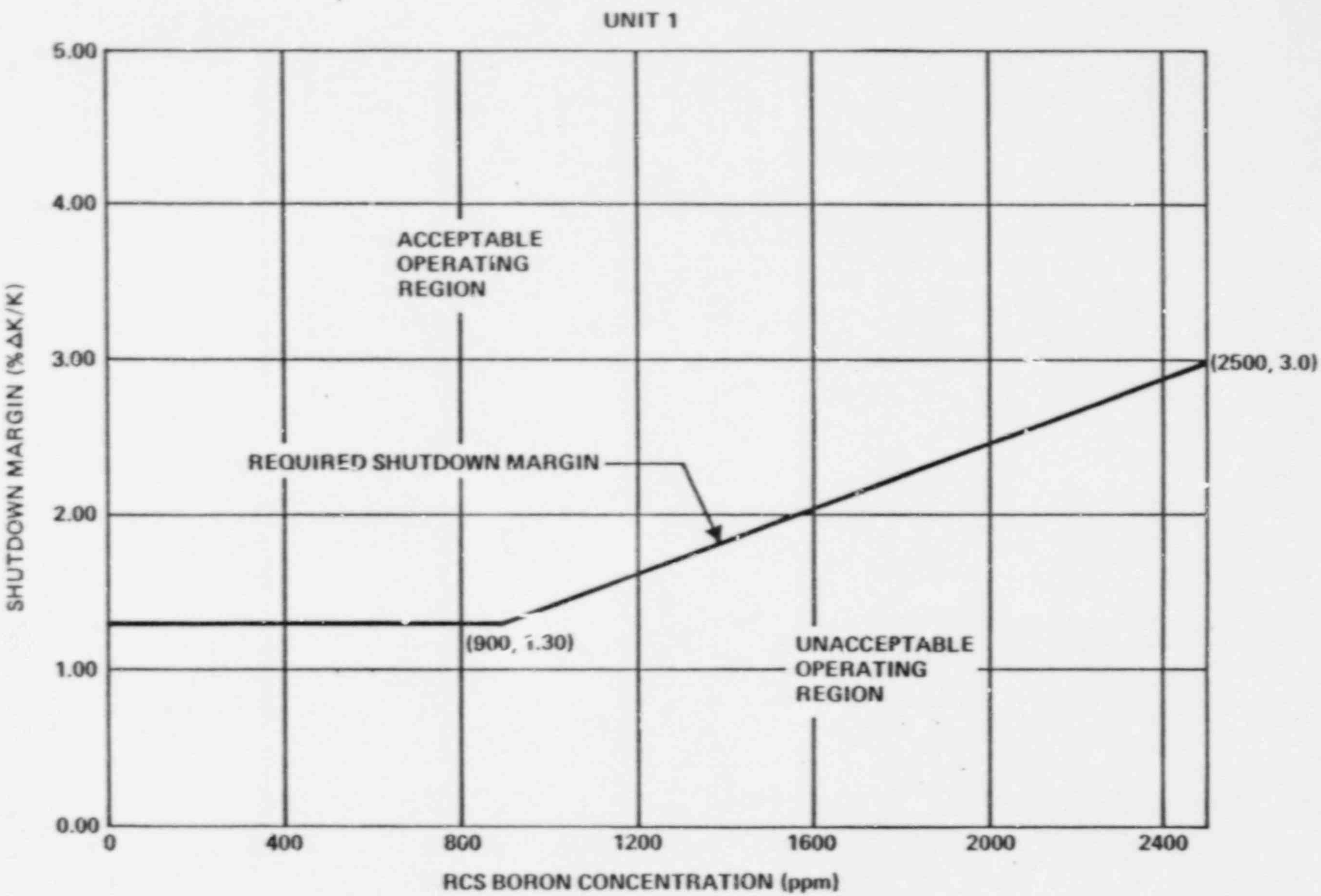

FIGURE 3.1-1a

REQUIRED SHUTDOWN MARGIN FOR MODES 3 AND 4 (MODE 4 WITH AT LEAST ONE RCP RUNNING) UNIT 1

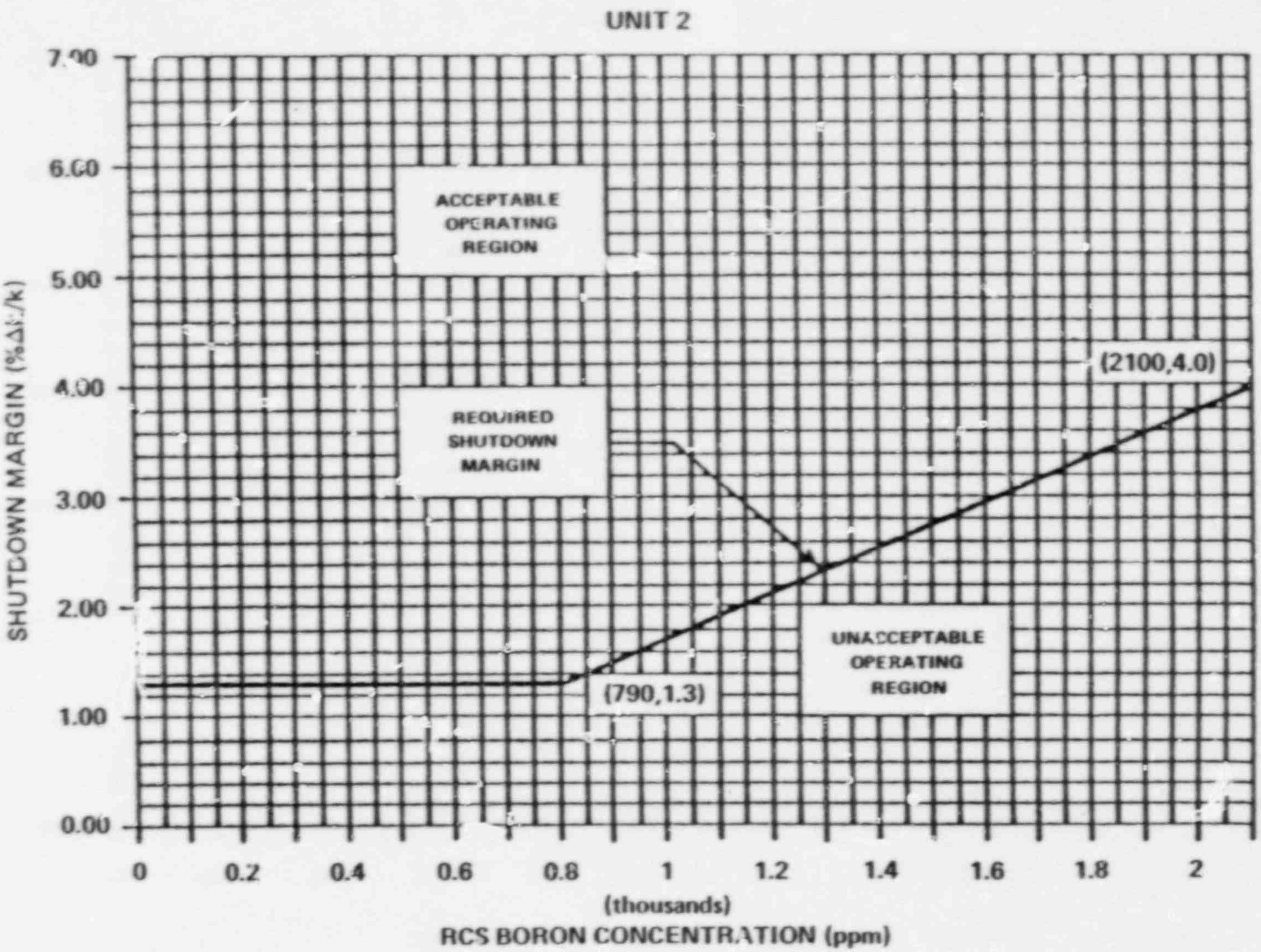

FIGURE 3.1-1b

REQUIRED SHUTDOWN MARGIN FOR MODES 3 AND 4 (MODE 4 WITH AT LEAST ONE REACTOR COOLANT PUMP RUNNING) UNIT 2

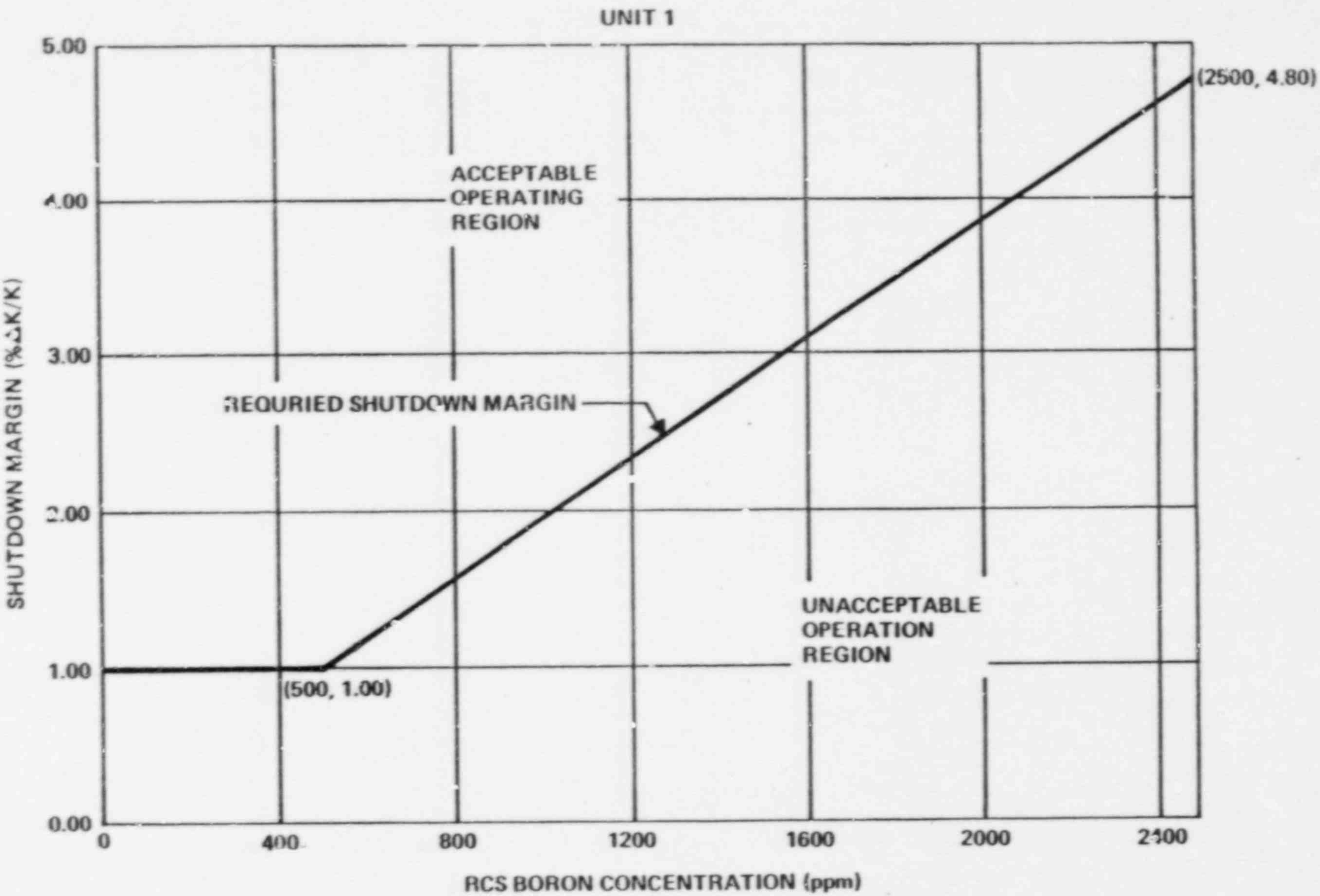

FIGURE 3.1-2a

REQUIRED SHUTDOWN MARGIN FOR MODE 5 (MODE 4 WITH NO RCPs RUNNING) UNIT 1

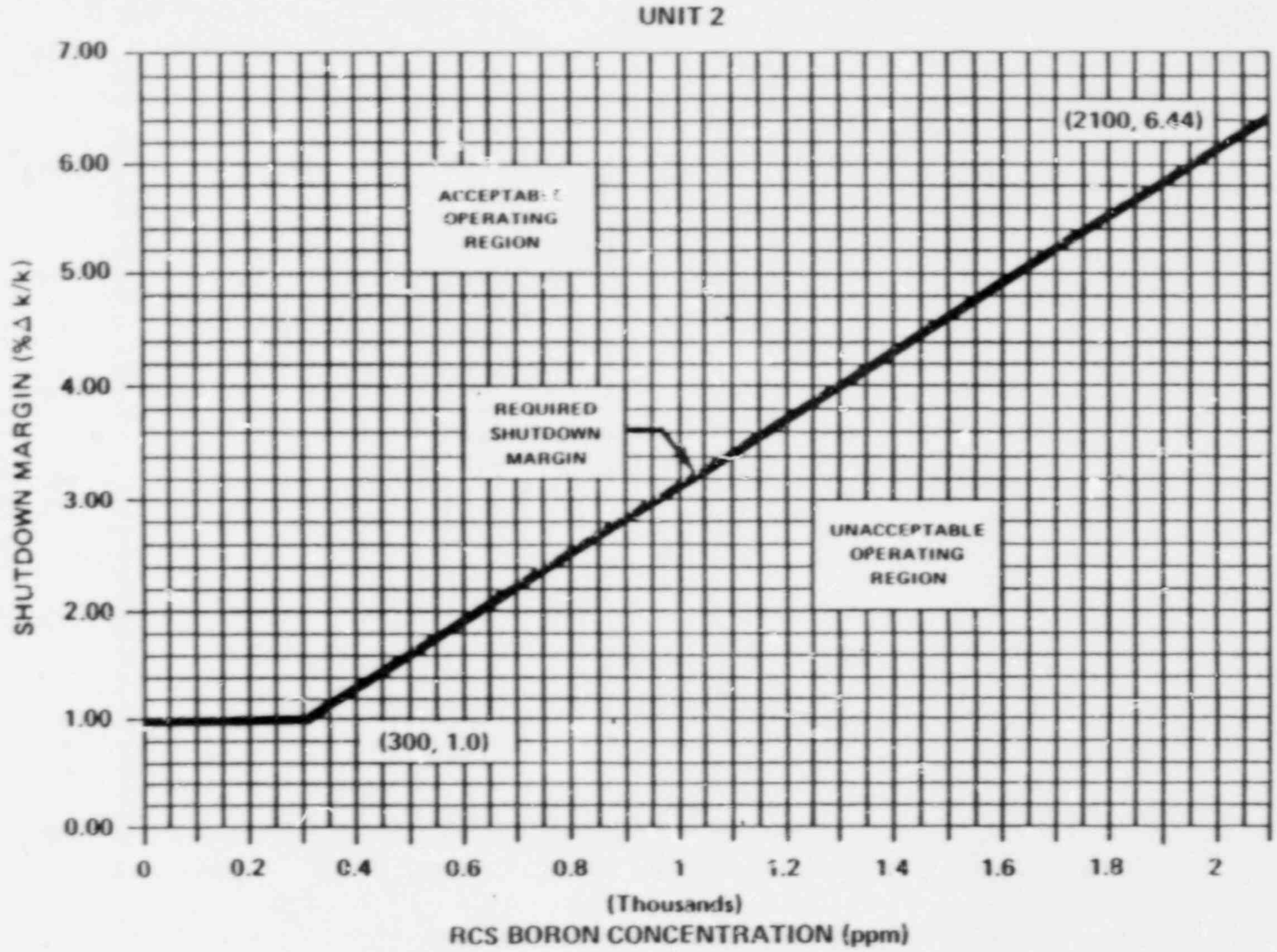

FIGURE 3.1-2b

REQUIRED SHUTDOWN MARGIN FOR MODE 5 (MODE 4 WITH NO RCPs RUNNING) UNIT 2

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

See insert AD.

~~a. Less positive than 0 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; and~~

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

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

ACTION:

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

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^{\circ}F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

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

   3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.8.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

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

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

**See Special Test Exceptions Specification 3.10.3.

INSERT AD

a. Unit 1:

Less positive than $+0.7 \times 10^{-4}$ $\Delta k/k/°F$ for the all rods withdrawn, beginning of cycle life (EOL) condition for power levels up to 70-percent RATED THERMAL POWER with a linear ramp to 0 $\Delta k/k/°F$ at 100-percent RATED THERMAL POWER.

Unit 2:

Less positive than 0 $\Delta k/k/°F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.

b. Unit 1:

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

Unit 2:.

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

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (TI-0412, TI-0422, TI-0432, TI-0442) (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2.**

#

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442) is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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

**See Special Test Exceptions Specification 3.10.3.

#

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid storage tank via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or

b. A flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days when the boric acid storage tank is a required water source, verify that the applicable portions of the auxiliary building (TISL 12410 or TISL 12411, TISL 12412 or TISL 12413, ~~TISL 12414~~ or TISL 12415, TISL 12416 or TISL 12417, TISL 20900 or TISL 20901, TISL 20902 or TISL 20903, and TISL 20904 or TISL 20905) are ≥72°F , and the portions of the flow path for which ambient temperature indication are not provided are ≥ 65°F, and

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

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

See insert AE.

~~1) A minimum contained borated water volume of 70832 gallons (5% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),~~

~~2) A boron concentration between 2000 ppm and 2200 ppm, and~~

3) A minimum solution temperature of 54°F (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 50°F.

INSERT AE

1) A minimim contained borated water volume of:

   Unit 1 - 99,404 gallons (9 percent of instrument span)
   Unit 2 - 70,832 gallons (5 percent of instrument span)

   (LI-0990A & B, LI-0991A & B, LI-0992A, LI-0993A)

2) A boron concentration between:

   Unit 1 - 2400 ppm and 2600 ppm
   Unit 2 - 2000 ppm and 2200 ppm

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

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

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

See insert AF.

~~2) A boron concentration between 2000 ppm and 2100 ppm,~~

3) A minimum solution temperature of 54°F, and

4) A maximum solution temperature of 116°F (TI-10982).

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

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

INSERT AF

2) A boron concentration between:

Unit 1 - 2400 ppm and 2600 ppm
Unit 2 - 2000 ppm and 2100 ppm

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE CONTROL OR SHUTDOWN ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Decrease in Reactor Coolant Inventory

- Inadvertent Opening of a Pressurizer Safety or Relief Valve
- Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundry That Penetrate Containment
- Loss-of-Coolant-Accidents

Increase in Heat Removal by the Secondary System (Steam System Piping Rupture)

Spectrum of Rod Cluster Control Assembly Ejection Accidents.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3,* # ** 4,* # ** and 5,* # **

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel, at least once per 18 months.

*With the Reactor Trip System breakers in the closed position.

** See Special Test Exceptions Specification 3.10.5.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall ~~be fully~~ withdrawn.

APPLICABILITY: MODES 1* and 2,* ** #

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-3.

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 figure, 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.
#

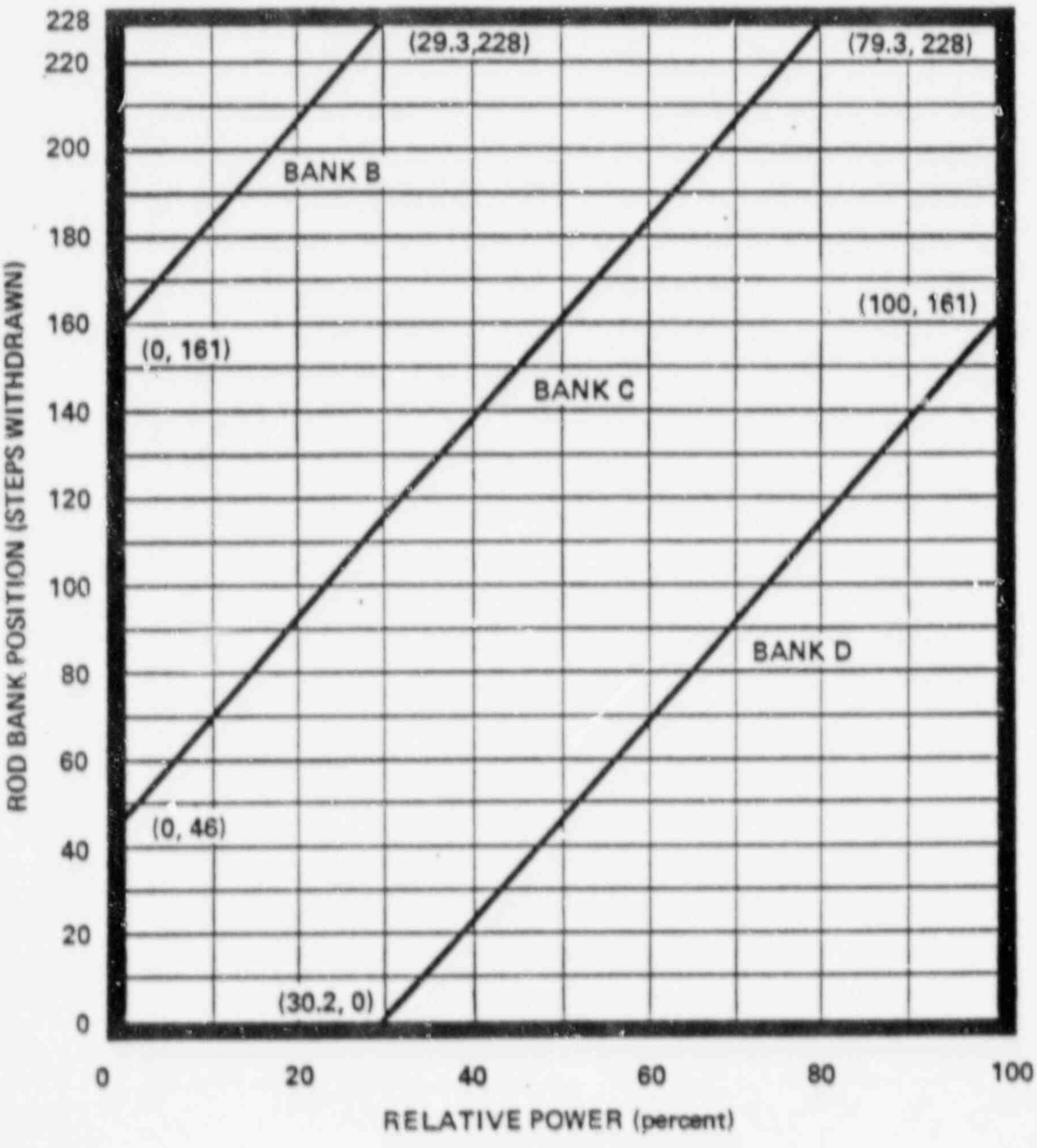

FIGURE 3.1-3

ROD BANK INSPECTION LIMITS VERSUS THERMAL POWER

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) 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 3.2-1 and the cumulative 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 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 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 RATED 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 Exceptions Specification 3.10.2.

** # Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

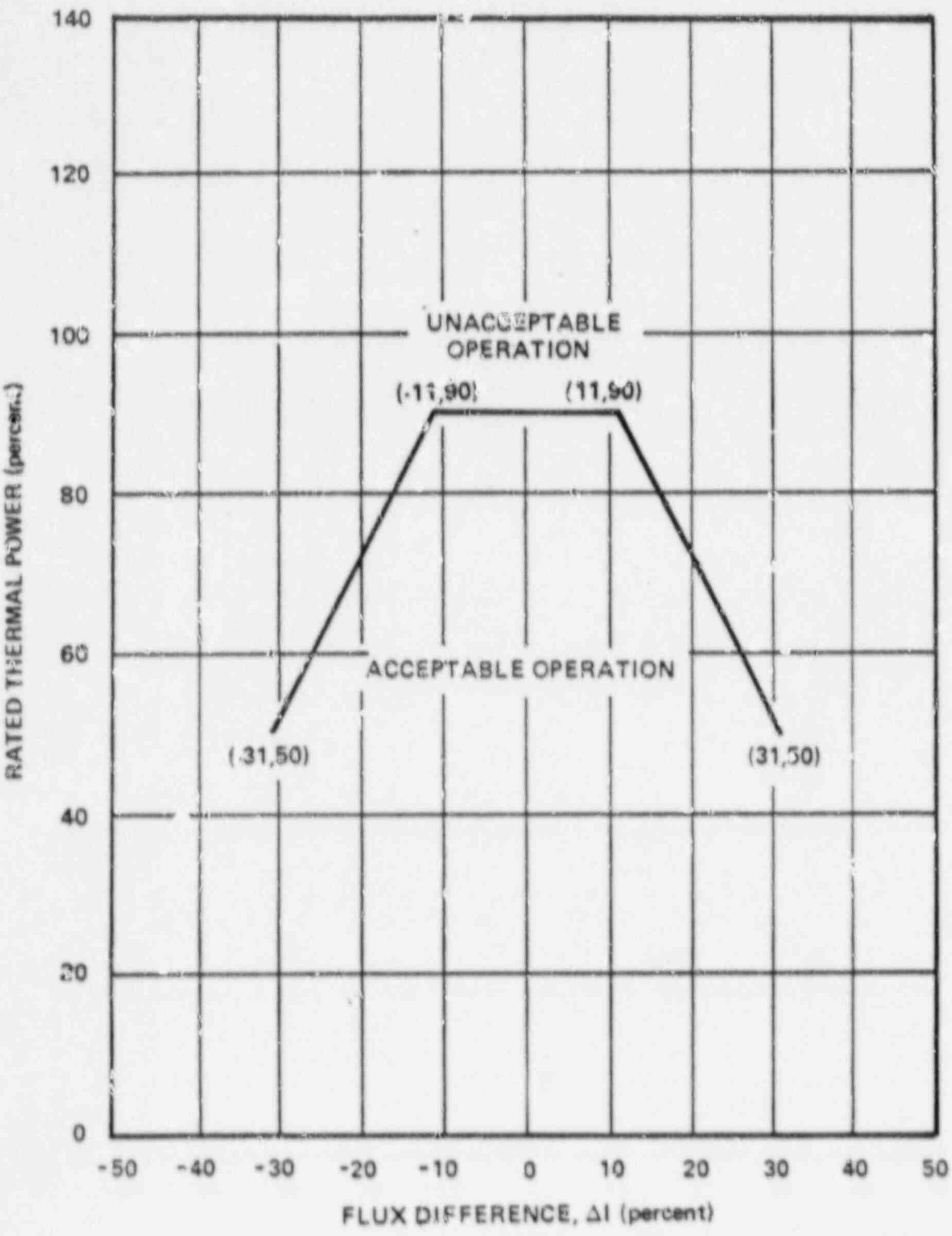

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

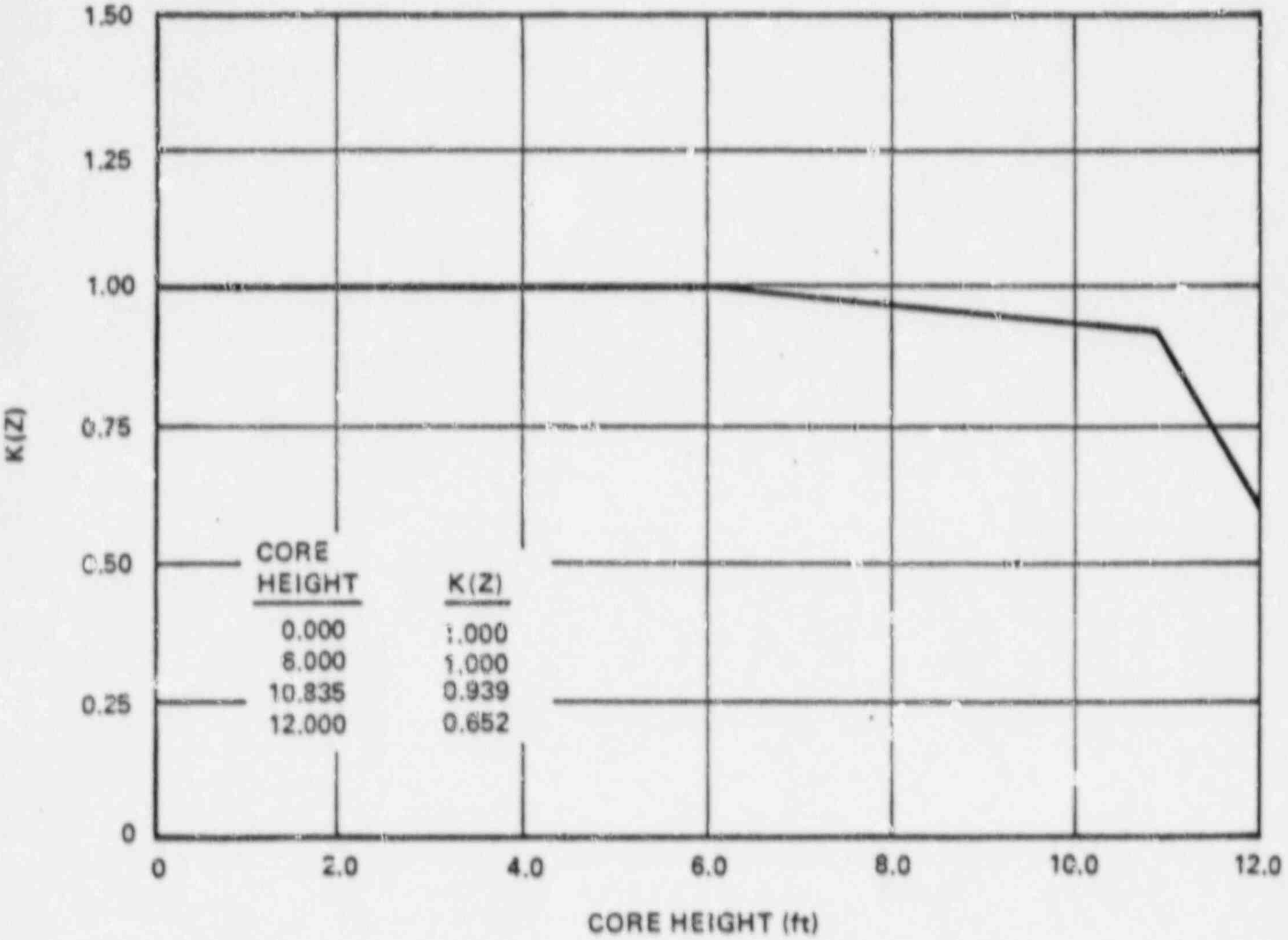

FIGURE 3.2-2

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Overtemperature ΔT (TDI-0411C, TDI-0421C, TDI-0431C, TDI-0441C)	4	2	3	1, 2	6^b
8. Overpower ΔT (TDI-0411B, TDI-0421B, TDI-0431B, TDI-0441B)	4	2	3	1, 2	6^b
9. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	4	2	3	1^f	6^b
10. Pressurizer Pressure--High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	4	2	3	1, 2	6^b
11. Pressurizer Water Level--High* (LI-0459A, LI-0460A, LI-0461A)	3	2	2	1^f	6^b
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1^h	6^b

(LOOP 1	LOOP 2	LOOP 3	LOOP 4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446)

*See Specification 3.3.3.6

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

b The provisions of Specification 3.0.4 are not applicable.

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

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

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

f Above the Power Reactor Trip Block) Setpoint.

g The applicab. Modes and Action Statement for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

h Above the P-8 (Single Loop Loss of Flow) Setpoint.

i Trip logic consists of under voltage/under frequency for Reactor Coolant Pumps 1 or 2 and 3 or 4.

j The Source Range High Flux at Shutdown Alarm may be blocked during reactor startup in accordance with approved procedures.

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 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 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.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Overpower ΔT (TDI-0411B, TDI-0421B, TDI-0431B, TDI-0441B)	S	R	Q(17)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	S	R	Q(17)	N.A.	N.A.	1[e]
10. Pressurizer Pressure--High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	S	R	Q(17)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High* (LI-0459A, LI-0460A, LI-0461A)	S	R	Q(17)	N.A.	N.A.	1[e]
12. Reactor Coolant Flow--Low	S	R	Q(17)	N.A.	N.A.	1[e]

(LOOP1	LOOP2	LOOP3	LOOP4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446)

*See Specification 4.3.3.6

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low* (LOOP1: LI-0517, LI-0518, LI-0519, LI-0551; LOOP2: LI-0527, LI-0526, LI-0529, LI-0552; LOOP3: LI-0537, LI-0538, LI-0539, LI-0553; LOOP4: LI-0547, LI-0548, LI-0549, LI-0554)	S	R	Q(17, 18)	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1^e
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1^e
16. Turbine Trip						
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	R	S/U,(1, 10)	N.A.	N.A.	1^b
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1^b
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6 (NI-0035B,D&E, NI-0036B,D&G)	N.A.	R(4)	R	N.A.	N.A.	2^c

*See Specification 4.3.3.6

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

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

e Above P-7 (Low Power Reactor Trip Block) Setpoint.

(1) If not performed in previous 31 days.

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

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

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

(5) Detector plateau curves shall be obtained, and evaluated. 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. This is the determination of the response of the excore power range detectors to the incore measured axial power distribution to generate setpoints for the CHANNEL CALIBRATION. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

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

(8) Not used.

(9) Quarterly surveillance in MODES 3^a, 4^a, and 5^a 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 window. Quarterly surveillance shall include verification of the Source Range High Flux at Shutdown Alarm Setpoint of less than or equal to ~~3.16~~ 2.3 (Unit 1), 3.16 (Unit 2) times background.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the Undervoltage and Shunt trip of the Reactor Trip Breaker.

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

(13) Not used.

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

(16) Automatic undervoltage trip.

Close up 1 line

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

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

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 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3 and with response times within their limit value.

APPLICABILITY: As shown in Table 3.3-2.

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 3.3-3, 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 3.3-3, either:

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3, 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 3.3-2 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 3.3-3 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 3.3-3 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-3 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

TABLE 3.3-2 (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)					
c. Containment Pressure--High-2* (PI-0934, PI-0935, PI-0936)	3	2	2	1, 2[f], 3[f]	15[d]
d. Steam Line Pressure--Low*	3/steam line	2/steam line any steam line	2/steam line	1, 2[f], 3[a,f]	15[d]
(LOOP1 PI-0514A,B&C PI-0515A PI-0516A; LOOP2 PI-0524A&B PI-0525A PI-0526A; LOOP3 PI-0534A&B PI-0535A PI-0536A; LOOP4 PI-0544A,B&C, PI-0545A, PI-0546A)					
e. Steam Line Pressure - Negative Rate--High **	3/steam line	2/steam line any steam line	2/steam line	3[b,f]	15[d]
(LOOP1 PI-0514A,B&C PI-0515A PI-0516A; LOOP2 PI-0524A&B PI-0525A PI-0526A; LOOP3 PI-0534A&B PI-0535A PI-0536A; LOOP4 PI-0544A,B&C PI-0545A PI-0546A)					
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	25
b. Low RCS Tavg Coincident with Reactor Trip**					
1. Low RCS T_{avg}	4	2	3	1, 2	20[d]
2. Reactor Trip, P-4	See Functional Unit 9b for P-4 requirements.				

* [illegible]

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip and Feedwater Isolation (Continued)					
c. Steam Generator Water Level--High-High (P-14)*	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm.gen. in each operating stm. gen.	1, 2	20[d]
(LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)					
d. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22

* See Specification 3.3.3.6.

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

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

b 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.

c During movement of irradiated fuel or movement of loads over irradiated fuel within containment.

d The provisions of Specification 3.0.4 are not applicable.

e During movement of irradiated fuel or movement of loads over irradiated fuel.

f Not applicable if one main steam isolation valve and associated bypass isolation valve per steamline is closed.

g Containment Ventilation Radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

h Manual initiation of Auxiliary Feedwater is accomplished via the pump handswitches.

either

i Whenever irradiated fuel is in the storage pool.

j For actions associated with inoperable instrumentation, follow actions specified in Specification 3.9.12.

ACTION STATEMENTS

ACTION 14 - 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 - 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 the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9 (Mode 6).

TABLE 3.3-3 (Continued)

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 (PI-0934, PI-0935, PI-0936)	3.1	0.71	1.67	≤14.5 psig	≤15.4 psig
d. Steam Line Pressure--Low	13.0	10.71	1.67	≥585 psig*	≥570 psig

(LOOP1	LOOP2	LOOP3	LOOP4
PI-0514A,B&C	PI-0524A&B	PI-0534A&B	PI-0544A,B&C
PI-0515A	PI-0525A	PI-0535A	PI-0545A
PI-0516A	PI-0526A	PI-0536A	PI-0546A)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
e. Steam Line Pressure - Negative Rate--High	3.0	0.50	0	≤100 psi**	≤125 psi

(LOOP1	LOOP2	LOOP3	LOOP4
PI-0514A,B&C	PI-0524A&B	PI-0534A&B	PI-0544A,B&C
PI-0515A	PI-0525A	PI-0535A	PI-0545A
PI-0516A	PI-0526A	PI-0536A	PI-0546A)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Low RCS Tavg Coincident with Reactor Trip##					
1. Low RCS T_{avg}	4.0	0.82	0.87	≥564°F	≥561.5°F
2. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (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 (Continued)					
c. Steam Generator Water Level--High-High (P-14)	5.1	2.18	1.67	≤78.0% of narrow range instrument span.	≤79.9% of narrow range instrument span.

(LOOP1	LOOP2	LOOP3	LOOP4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Safety Injection	See Functional Unit 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--Low-Low					

(LOOP1	LOOP2	LOOP3	LOOP4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Start Motor-Driven Pumps	18.5	17.18	1.67	≥18.5% of narrow range instrument span.	≥17.8% of narrow range instrument span.
2. Start Turbine-Driven Pump	18.5	17.18	1.67	≥18.5% of narrow range instrument span.	≥17.8% of narrow range instrument span.

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Lc are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ 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 greater than or equal to 50 seconds. CHANNEL CALIBRATI shall ensure that this time constant is adjusted to this value.

***Until resolution of the Veritrak transmitter uncertainty issue this setpoint will be set at ≥1885 psig.

#Until resolution of the Veritrak transmitter uncertainty issue the setpoint will be set at ≥22.5% (Unit 1) and ≥ 23.5% (Unit 2) of narrow range instrument span.

##Feedwater isolation only. Turbine trip occurs on reactor trip.

[a]During refueling operations.

[b]During power operation. This is an initial setpoint only. The trip setpoint will be set at 50 times background level. Background level should be determined at or near the end of the first fuel cycle.

[c]Setpoints will not exceed the limits of Specification 3.11.2.1.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES 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. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Low RCS Tavg Coincident with Reactor Trip*								
1. Low RCS T_{avg}	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
2. Reactor Trip, P-4		See Functional Unit 9b for P-4 Surveillance requirements.						
c. Steam Generator Water Level-High-High (P-14)**	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2

(LOOP1	LOOP2	LOOP3	LOOP4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554)

d. Safety Injection See Functional Unit 1 above for all Safety Injection Surveillance Requirements.

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*Feedwater isolation only. Turbine trip occurs on reactor trip.

**See Specification 4.3.3.6

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
10. Control Room Ventilation Emergency Mode Actuation (Continued)								
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements.							
d. Intake Radiogas Monitor (RE-12116, RE-12117)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 5#, 6#
11. Fuel Handling Building Post Accident Ventilation Actuation (Common System)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	(2)
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532 A&B ARE-2533 A&B)	S	R	M	N.A.	N.A.	N.A.	N.A.	(2)
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A	N.A.	M(1)	N.A.	N.A.	(2)

either

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Whenever irradiated fuel is in the storage pool.
During movement of irradiated fuel or movement of loads over irridated fuel.

SEISMIC INSTRUMENTATION (Common System)

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments which is accessible during power operations and which is actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 15 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

Each of the above seismic monitoring instruments which is actuated during a seismic event greater than or equal to 0.01 g but is not accessible during power operation shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed the next time Unit 1 enters MODE 5 or below. A supplemental report shall then be prepared and submitted to the Commission with 14 days pursuant to Specification 6.8.2 describing the additional data from these instruments.

TABLE 3.3-5

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE	INSTRUME[illegible] TAG NUMBER
1. Triaxial Time-History Accelerographs			
a. Free Field (500 ft from containment)	-1 to 1 g	1	AXT-1990[illegible]
b. Unit 1 Containment Gallery (basemat)	-1 to 1 g	1	AXT-1990[illegible]
c. Unit 1 Containment Operating Floor	-1 to 1 g	1	AXT-1990[illegible]
d. Auxiliary Building Basemat	-1 to 1 g	1	AXT-19906
e. Unit 1 Containment Pressurizer Support	-1 to 1 g	1	AXT-1990[illegible]
f. Auxiliary Building Level 1	-1 to 1 g	1	AXT-1990[illegible]
2. Triaxial Peak Accelerographs			
a. Unit 1 Reactor Coolant Pump Motor (210 ft)	±10gHoriz/±5gVert	1	AXR-1991[illegible]
b. Unit 1 Steam Generator (185 ft)	±2gHoriz/±5gVert	1	AXR-1991[illegible]
c. Unit 1 NSCW Piping Outside Aux Bldg (220 ft)	-10g to +10g	1	AXR-1991[illegible]
3. Triaxial Seismic Switch			
a. Unit 1 Containment Tendon Gallery (basemat)	**	1*	AXSH-199[illegible]
4. Triaxial Reponse-Spectrum Analyzer			
a. Control Room	Input: -1 to 1g Output: 0.03g to 9.99g	1*	AXA-19930
5. Triaxial Seismic Triggers			
a. Unit 1 Containment Tendon Gallery (basemat)	***	1*	AXSH-1992
b. Unit 1 Containment Operating Floor	***	1*	AXSH-1992

*With reactor control room indication.

**Triaxial seismic switch is set at the OBE acceleration level of 0.17g horizontal and 0.23g vertical.

***Triaxial seismic trigger is set at 0.01g all axes.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	ACCESS DURI MODE
1. Triaxial Time-History Accelerographs				
a. Free Field (500 ft from containment)	M	R	SA	All
b. Unit 1 Containment Gallery (basemat)	M	R	SA	All
c. Unit 1 Containment Operating Floor	M	R	SA	All
d. Auxiliary Building Basemat	M	R	SA	All
e. Unit 1 Containment Pressurizer Support	M	R	SA	5,
f. Auxiliary Building Level 1	M	R	SA	All
2. Triaxial Peak Accelerographs				
a. Unit 1 Reactor Coolant Pump Motor (210 ft)	N.A.	R	N.A.	5,
b. Unit 1 Steam Generator (185 ft)	N.A.	R	N.A.	5,
c. Unit 1 NSCW Piping Outside Aux. Bldg. (220 ft)	N.A.	R	N.A.	All
3. Triaxial Seismic Switches				
a. Unit 1 Containment Tendon Gallery (basemat)	M	R	SA	All
4. Triaxial Response-Spectrum Analyzer				
a. Control Room*	M	R	N.A.	All
5. Triaxial Seismic Triggers				
a. Unit 1 Containment Tendon Gallery (basemat)	M	R	SA	All
b. Unit 1 Containment Operating Floor	M	R	SA	All

*With reactor control room indications.

TABLE 3.3-6

METEOROLOGICAL MONITORING INSTRUMENTATION*

INSTRUMENT		LOCATION	MINIMUM OPERABLE
1.	Wind Speed		
	a. Lower, Primary Tower	Nominal Elev. 10 m	1
	b. Upper, Primary Tower	Nominal Elev. 60 m	1
2.	Wind Direction		
	a. Lower, Primary Tower	Nominal Elev. 10 m	1
	b. Upper, Primary Tower	Nominal Elev. 60 m	1
3.	Air Temperature - ΔT		
	a. ΔT, Primary Tower	Nominal Elev. 10m-60m	1

E

*This instrumentation is common to Units 1 and 2.

TABLE 3.3-7

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

INSTRUMENT Function	READOUT[1] LOCATION	CHANNELS AVAILABLE	MINIMUM CHANNELS OPERABLE
1. Source Range Neutron Flux	A	1 (NI-31E)	1
2. Extended Range Neutron Flux	B	1 (NI-13135 C&D)	1
3. RCS LP Cold Leg Temperature	A, B	1/Loop (Loop 1 TI-0413D, Panel A) (Loop 2 TI-0423D, Panel B) (Loop 3 TI-0433D, Panel B) (Loop 4 TI-0443D, Panel A)	1/Loop
4. RCS Hot Leg Temperature	A	2 (Loop 1 TI-0413C Loop 4 TI-0443C)	2
5. Core Exit Thermocouples	B	2 (Loop 2 Core Quandrant TI-10055 Loop 3 Core Quandrant TI-10056)	2
6. RCS Wide Range Pressure	A, B	2 (PI-405A, Panel A) (PI-403A, Panel B)	2
7. Steam Generator Level Wide Range	A, B	1/Loop (Loop 1 LI-501B, Panel A) (Loop 2 LI-502B, Panel B) (Loop 3 LI-503B, Panel B) (Loop 4 LI-504B, Panel A)	1/Loop
8. Pressurizer Level	A, B	2 (LI-459C, Panel A) (LI-460C, Panel B)	2
9. RWST Level	L	1 (LI-0990C)	1[3]

TABLE 3.3-7 (Continued)

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

INSTRUMENT Function	READOUT[1] LOCATION	CHANNELS AVAILABLE	MINIMUM CHANNELS OPERABLE
10. BAST Level	L	1 (PI-10115[2])	1[3]
11. CST Level	L	2 (Tank 1 LI-5100) (Tank 2 LI-5115)	2[3]
12. Auxiliary Feedwater Flow	A, B	1/LOOP (LOOP1 FI-5152B, Panel A) (LOOP2 FI-5151B, Panel B) (LOOP3 FI-5153B, Panel B) (LOOP4 FI-5150B, Panel A)	1/LOOP
13. Steam Generator Pressure	A, B	1/LOOP (LOOP1 PI-0514C, Panel A) (LOOP2 PI-0525B, Panel B) (LOOP3 PI-0535B, Panel B) (LOOP4 PI-0544C, Panel A)	1/LOOP

[1] A - Remote Shutdown Panel PSDA
B - Remote Shutdown Panel PSDB
L - Local Indication

[2] Graph will be provided to determine level from pressure reading

[3] Alternate local level indication may be established to fulfill the minimum channels OPERABLE.

TABLE 3.3-8

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Reactor Coolant Pressure (Wide Range) (Loop 408, 418, 428, & 438)	4	1	35
2. Reactor Coolant System T_{hot} (Wide Range) (Loop 413A, 423A, 433A & 443A)	1/loop	1/loop	32
3. Reactor Coolant System T_{cold} (Wide Range) (Loop 413B, 423B, 433B & 443B)	1/loop	1/loop	32
4. SG Water Level (Wide Range) (Loop 501, 502, 503 & 504)	1/SG	1/SG	32
5. SG Water Level (Narrow Range) (Loop 517, 518, 519, 527, 528, 529, 537, 538, 539, 547, 548, 549, 551, 552, 553, 554)	4/SG	1/SG	35
6. Pressurizer Level (Loop 459, 460, 461)	3	1	30
7. Containment Pressure (Loop 934, 935, 936, 937)	4	1	35
8. Steamline Pressure (Loop 514, 515, 516, 524, 525, 526, 534, 535, 536, 544, 545 & 546)	3/stm. line	1/stm. line	30
9. RWST Level (Loop 990, 991, 992 & 993)	4	1	35
10. Containment Normal Sumps Level (Narrow Range) (Loop 7777 & 7789)	2	1	31
11. Containment Water Level (Wide Range) (Loop 0764 & 0765)	2	1	31

TABLE 3.3-8 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
12.	Condensate Storage Tank Level (Loop 5101, 5111, 5104 & 5116)	2/tank	1/tank	31
13.	Auxiliary Feedwater Flow (Loop 5152, 15152, 5153, 15153, 5151, 15151, 5150 & 15150)	2/feed line	1/feed line	31
14.	Containment Radiation Level (High Range) (Loop 0005 & 0006)	2	1	33
15.	Steamline Radiation Monitor (Loop 13119, 13120, 13121 & 13122)	1/stm. line	1/stm. line	33
16.	Core Exit Thermocouples	4/quad/train	2/quad/train	30
17.	Reactor Coolant System Subcooling	2	1	31
18.	Neutron Flux (Extended Range) (Loop 13135A & 13135B)	2	1	31
19.	RVLIS	2	1	34
20.	Containment Hydrogen Concentration (Loop 12979 & 12980)	2	1	31
21.	Containment Pressure (Extended Range) (Loop 10942 & 10943)	2	1	31
22.	Containment Isolation Valve Position Indication*	1/valve	1/valve	36

*Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A or containment ventilation isolation signals).

TABLE 3.3-8 (Continued)

ACTION STATEMENTS

ACTION 30 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels two less then the Total Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN with the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 31 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 32 - With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.4 are not applicable.

ACTION 33 - With the number of OPERABLE channels less than the minimum channels OPERABLE requirement, initiate the alternate method of monitoring the parameter within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.8.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 34 - With the number of OPERABLE channels less than the required number of channels or the minimum channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status as per Action 31a or b as applicable if repair is feasible during plant operation. If repair is not feasible, prepare and submit a Special Report to the Commission, pursuant to Specification 6.8.2 within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.*

ACTION 35 - a. With the number of OPERABLE channels two less than the Total Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels three less than the Total Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status with 48 hours or be in HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 36 - With the number of OPERABLE channels less than the minimum channels OPERABLE requirements, comply with the provisions of Specification 3.6.3. for an inoperable containment isolation valve.

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*Action Statement 34 applies to the first fuel cycle only. Action Statement 31 is applicable thereafter.

TABLE 3.3-9

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RE-0018)	1	37
b. Steam Generator Blowdown Effluent Line (RE-0021)	1	38
c. Turbine Building (Floor Drains) Sumps Effluent Line (RE-0848)	1	38
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Nuclear Service Cooling Water System Effluent Line (RE-0020 A & B)	1	39
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (FT-0018)	1	40
b. Steam Generator Blowdown Effluent Line (FT-0021)	1	40
c. Flow to Blowdown Sump (AFQI-7620, FR-7620, pen 1) (Common)	1	40

TABLE 4.3-5 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line (FT-0018)	D(4)	N.A.	R	N.A.
b. Steam Generator Blowdown Effluent Line (FT-0021)	D(4)	N.A.	R	N.A.
c. Flow to Blowdown Sump (AFQI-7620, FR-7620 pen 1)	D(4)	N.A.	R	Q

TABLE 3.3-10

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS WASTE PROCESSING SYSTEM			
4 lines a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ARE-0014) Common	1	***	45
3 lines b. Effluent System Flow Rate Measuring Device (AFT-0014) Common	1	***	46
2. GASEOUS WASTE PROCESSING ~~SYSTEM~~ Explosive Gas Monitoring System			
a. Hydrogen Monitor	1/recombiner	**	50
b. Oxygen Monitor	2/recombiner	**	49
3. Condenser Air Ejector and Steam Packing Exhauster System			
a. Noble Gas Activity Monitor (RE-12839C)	1	***	47
b. Iodine Sampler (RE-12839B)	1	***	51
c. Particulate Sampler (RE-12839A)	1	***	51
d. Flow Rate Monitor (FT-12839) (FIS-12862)#	1	***	46
e. Sampler Flow Rate Monitor (FT-13211)	1	***	46

TABLE 3.3-10 (Continued)

TABLE NOTATIONS

* At all times.
** During GASEOUS WASTE PROCESSING SYSTEM operation.
***During radioactive releases via this pathway.
\# During Emergency Filtration.

ACTION STATEMENTS

ACTION 41-44 (Not Used)

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend containment PURGING of radioactive effluents via this pathway.

ACTION 49 - a. With the outlet oxygen monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours and the oxygen concentration remains less than 1 percent.

b. With the inlet oxygen monitor inoperable, operation may continue if the inlet hydrogen monitor is OPERABLE.

c. With both oxygen channels or both of the inlet oxygen and inlet hydrogen monitors inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remains less than 1 percent.

TABLE 4.3-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS WASTE PROCESSING SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RE-0014)	P	P	R(3)	Q(1)	c
b. Effluent System Flow Rate Measuring Device (AFT-0014)	P	N.A.	R	N.A.	c
2. GASEOUS WASTE PROCESSING SYSTEM Explosive Gas Monitoring System					
a. Hydrogen Monitors	D	N.A.	Q(4)	M	b
b. Oxygen Monitors	D	N.A.	Q(5)	M	b
3. Condenser Air Ejector and Steam Packing Exhauster System					
a. Noble Gas Activity Monitor (RE-12839C)	D	M	R(3)	Q(2)	c
b. Iodine Sampler (RE-12839B)	W(6)	N.A.	N.A.	N.A.	c
c. Particulate Sampler (RE-12839A)	W(6)	N.A.	N.A.	N.A.	c
d. Flow Rate Monitor (FT-12839)	D	N.A.	R	N.A.	c
e. Sampler Flow Rate Monitor (FI-1321[illegible])	D	N.A.	R	Q	c

TABLE 3.3-11

HIGH-ENERGY LINE BREAK INSTRUMENTATION

	Isolation Function	Instrument Channel	Minimum Channels Operable	Applicable Modes
1.	Electric Steam Boiler Isolation (Common Instrumentation)	ATE 19722A (RD52) ATE 19723A (RD52)	1	*
		ATE 19722B (RC41) ATE 19723B (RC41)	1	*
		ATE 19722C (RC65) ATE 19723C (RC64)	1	*
		ATE 19722D (RC67) ATE 19723D (RC66)	1	*
		AFT 19722 AFT 19723	1	*
		ATE 19722E (RC95) ATE 19723E (RC95)	1	*

	Isolation Function	Instrument Channel (Unit 1)	Instrument Channel (Unit 2)	Minimum Channels Operable	Applicable Modes
2.	Steam Generator Blowdown Line Isolation	TE 15212A (RB08) TE 15216A (RB08)	TE 15212A(RB131) TE 15216A(RB131)	1	1, 2, 3, 4
		TE 15212B (RC106) TE 15216B (RC106)	TE 15212B(RC103) TE 15216B(RC103)	1	1, 2, 3, 4
		TE 15212C (RC107) TE 15216C (RC107)	TE 15212C(RC101) TE 15216C(RC101)	1	1, 2, 3, 4
		TE 15212D (RC108) TE 15216D (RC108)	TE 15212D(RC102) TE 15216D(RC102)	1	1, 2, 3, 4
		FT 15212A (Loop 1) FT 15216A	FT 15212A(Loop 1) FT 15216A	1	1, 2, 3, 4
		FT 15212B (Loop 2) FT 15216B	FT 15212B(Loop 2) FT 15216B	1	1, 2, 3, 4
		FT 15212C (Loop 3) FT 15216C	FT 15212C(Loop 3) FT 15216C	1	1, 2, 3, 4
		FT 15212D (Loop 4) FT 15216D	FT 15212D(Loop 4) FT 15216D	1	1, 2, 3, 4
3.	Letdown Line Isolation	TE 15214A (A07) TE 15215A (A07)	TE 15214A(A100) TE 15215A(A100)	1	1, 2, 3, 4
		TE 15214B (A08) TE 15215B (A08)	TE 15214B(A101) TE 15215B(A101)	1	1, 2, 3, 4
		TE 15214C (A09) TE 15215C (A09)	TE 15214C(A103) TE 15215C(A103)	1	1, 2, 3, 4

*Required during all MODES when electric steam boiler is in operation.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops/trains listed below shall be OPERABLE and at least one of these loops/trains shall be in operation:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,**

e. RHR train A, and

f. RHR train B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops/trains OPERABLE, immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop/train is an RHR train, be in COLD SHUTDOWN within 24 hours.

b. With no loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop/train to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) train shall be OPERABLE and in operation*, and either:

a. One additional RHR train shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 17% of wide range (LI-0501, LI-0502, LI-0503, LI-0504).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR trains inoperable or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR train to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR train is OPERABLE and in operation.

***A reactor coolant pump shall not be started unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) trains shall be OPERABLE* and at least one RHR train shall be in operation.** Reactor Makeup Water Storage Tank (RMWST) discharge valves (1208-U4-175, 1208-U4-176, 1208-U4-177 and 1208-U4-183) shall be closed and secured in position.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR trains OPERABLE, immediately initiate corrective action to return the required RHR trains to OPERABLE status as soon as possible.

b. With no RHR train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train to operation.

c. With the Reactor Makeup Water Storage Tank (RMWST) discharge valves (1208-U4-175, 1208-U4-176, 1208-U4-177, and 1208-U4-183) not closed and secured in position, immediately close and secure in position the RMWST discharge valves.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

7

4.4.1.4.2.2 Valves 1208-U4-175, 1208-U4-176, 1208-U4-177, and 1208-U4-183 shall be verified closed and secured in position by mechanical stops at least once per 31 days.

---

*One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR train is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve, and

   1. With only one PORV OPERABLE, restore at least a total of two PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no PORVs OPERABLE, restore at least one PORV to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour (1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s) or close the PORV and remove power from its associated solenoid valve; and (2) apply ACTION b above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION.

*The provisions of this specification are not applicable to Unit 2 until initial entry of Unit 2 into MODE 2.

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.8.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a special report to the Commission pursuant to specification 6.8.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours;

b. Monitoring the containment normal sumps and reactor cavity sump inventory at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

See insert AG.

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once during each refueling outage (leak testing should be performed after all disturbances to the valves are complete, such as before reaching power operation during a refueling outage);

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve (leak testing should be performed after all disturbances to the valves are complete);

c. For systems rated at less than 50% of RCS design pressure, each time the valve has moved from its fully closed position except for valves HV-8701 A/B and HV-8702 A/B.

d. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months except for valves HV-8701 A/B and HV-8702 A/B.

e. As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

INSERT AG

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

	VALVE NUMBER	VALVE SIZE(in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE(gpm)
1.	HV-8701A	12	RHR Suction (gate valve)	5.0
2.	HV-8701B	12	RHR Suction (gate valve)	5.0
3.	HV-8702A	12	RHR Suction (gate valve)	5.0
4.	HV-8702B	12	RHR Suction (gate valve)	5.0
5.	1204-U4-120	2	SI-Hot Leg 2nd Isolation Valve	1.0
6.	1204-U4-121	2	SI-Hot Leg 2nd Isolation Valve	1.0
7.	1204-U4-122	2	SI-Hot Leg 2nd Isolation Valve	1.0
8.	1204-U4-123	2	SI-Hot Leg 2nd Isolation Valve	1.0
9.	1204-U6-079	10	Accumulator 2nd Isolation Valve	5.0
10.	1204-U6-080	10	Accumulator 2nd Isolation Valve	5.0
11.	1204-U6-081	10	Accumulator 2nd Isolation Valve	5.0
12.	1204-U6-082	10	Accumulator 2nd Isolation Valve	5.0
13.	1204-U6-083	10	Injection Line 1st Isolation Valve	5.0
14.	1204-U6-084	10	Injection Line 1st Isolation Valve	5.0
15.	1204-U6-085	10	Injection Line 1st Isolation Valve	5.0
16.	1204-U6-086	10	Injection Line 1st Isolation Valve	5.0
17.	1204-U6-124	6	SI-Hot Leg 1st Isolation Valve	3.0
18.	1204-U6-125	6	SI-Hot Leg 1st Isolation Valve	3.0
19.	1204-U6-126	6	SI-Hot Leg 1st Isolation Valve	3.0
20.	1204-U6-127	6	SI-Hot Leg 1st Isolation Valve	3.0
21.	1204-U6-128	8	RHR-Hot Leg 2nd Isolation Valve	4.0
22.	1204-U6-129	8	RHR-Hot Leg 2nd Isolation Valve	4.0
23.	1204-U4-143	2	SI-Cold Leg 2nd Isolation Valve	1.0
24.	1204-U4-144	2	SI-Cold Leg 2nd Isolation Valve	1.0
25.	1204-U4-145	2	SI-Cold Leg 2nd Isolation Valve	1.0
26.	1204-U4-146	2	SI-Cold Leg 2nd Isolation Valve	1.0
27.	1204-U6-147	6	RHR Cold Leg 2nd Isolation Valve	3.0
28.	1204-U6-148	6	RHR Cold Leg 2nd Isolation Valve	3.0
29.	1204-U6-149	6	RHR Cold Leg 2nd Isolation Valve	3.0
30.	1204-U6-150	6	RHR Cold Leg 2nd Isolation Valve	3.0

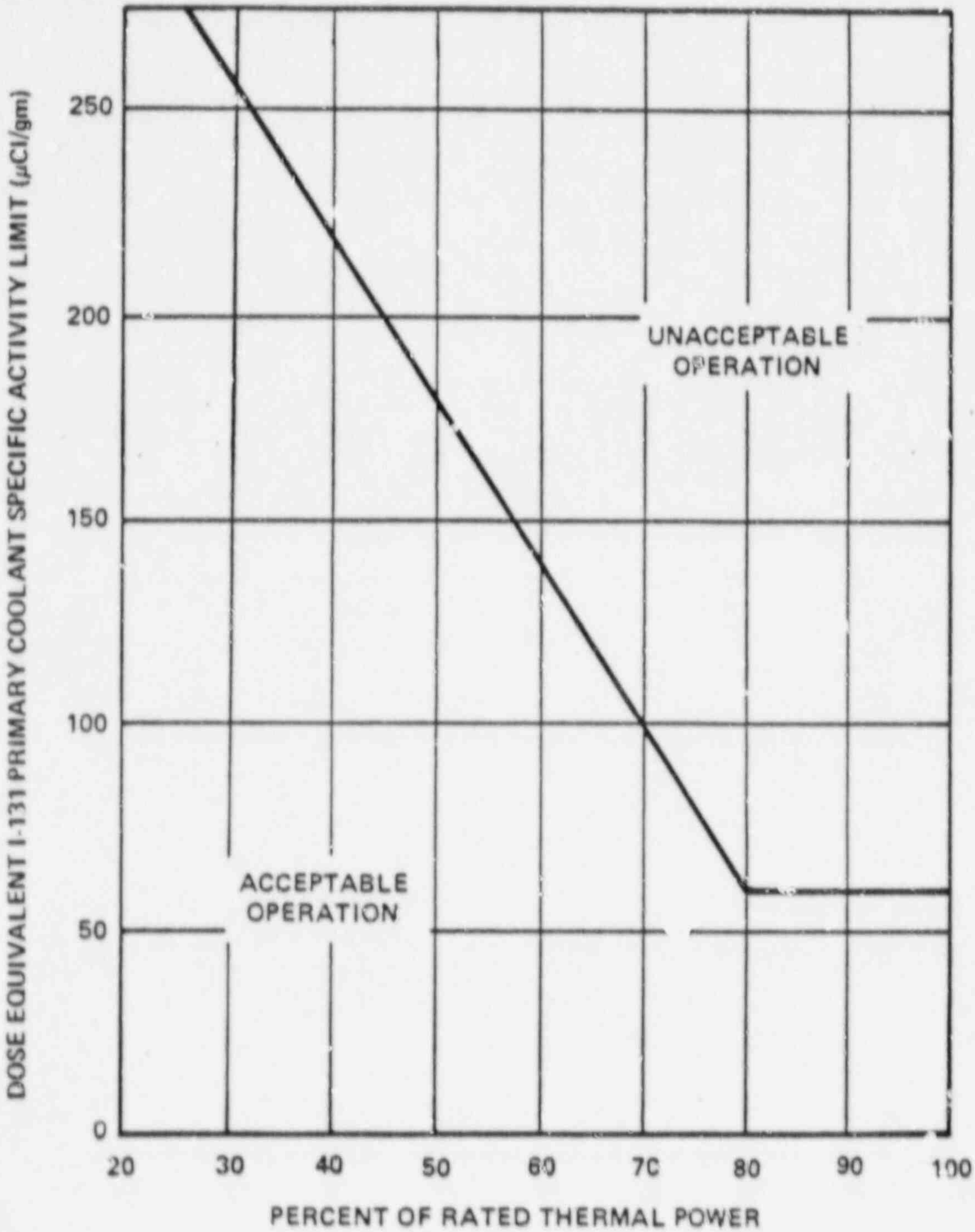


FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY $\geq$ 1 μCi/gram DOSE EQUIVALENT I-131

3/4.4.9 PRESSURE/TEMPERATURE LIMITS ~~[TO BE REVISED TO REFLECT UNIT 2]~~

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on ~~Figures 3.4-2 and 3.4-3~~ during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

See insert AH.

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

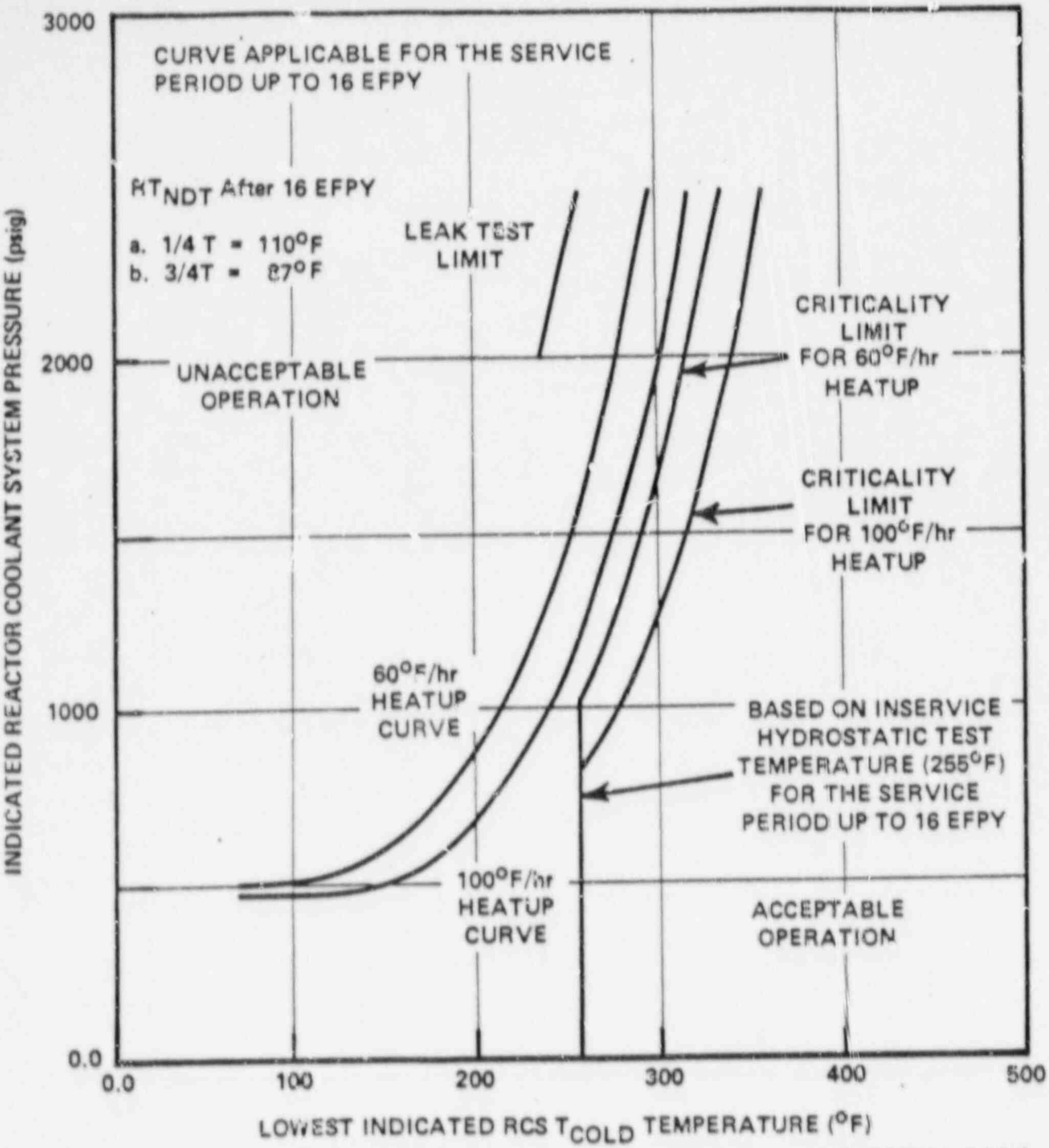
SURVEILLANCE REQUIREMENTS

4.4.9.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

FIGURE 3.4-2

INSERT AH

Figures 3.4-2a (Unit 1) and 3.4-3a (Unit 1), Figures 3.4-2b (Unit 2) and 3.4-3b (Unit 2)



MATERIAL BASIS

Copper Content: Assumed = 10 Wt %
(Actual = 06 Wt %)

RT_{NDT} Initial: Assumed = 40° F
(Actual = 30° F)

RT_{NDT} After 16 EFPY @ 1/4T = 110° F
@ 3/4T = 87° F

FIGURE 3.4-2a

UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 16 EFPY

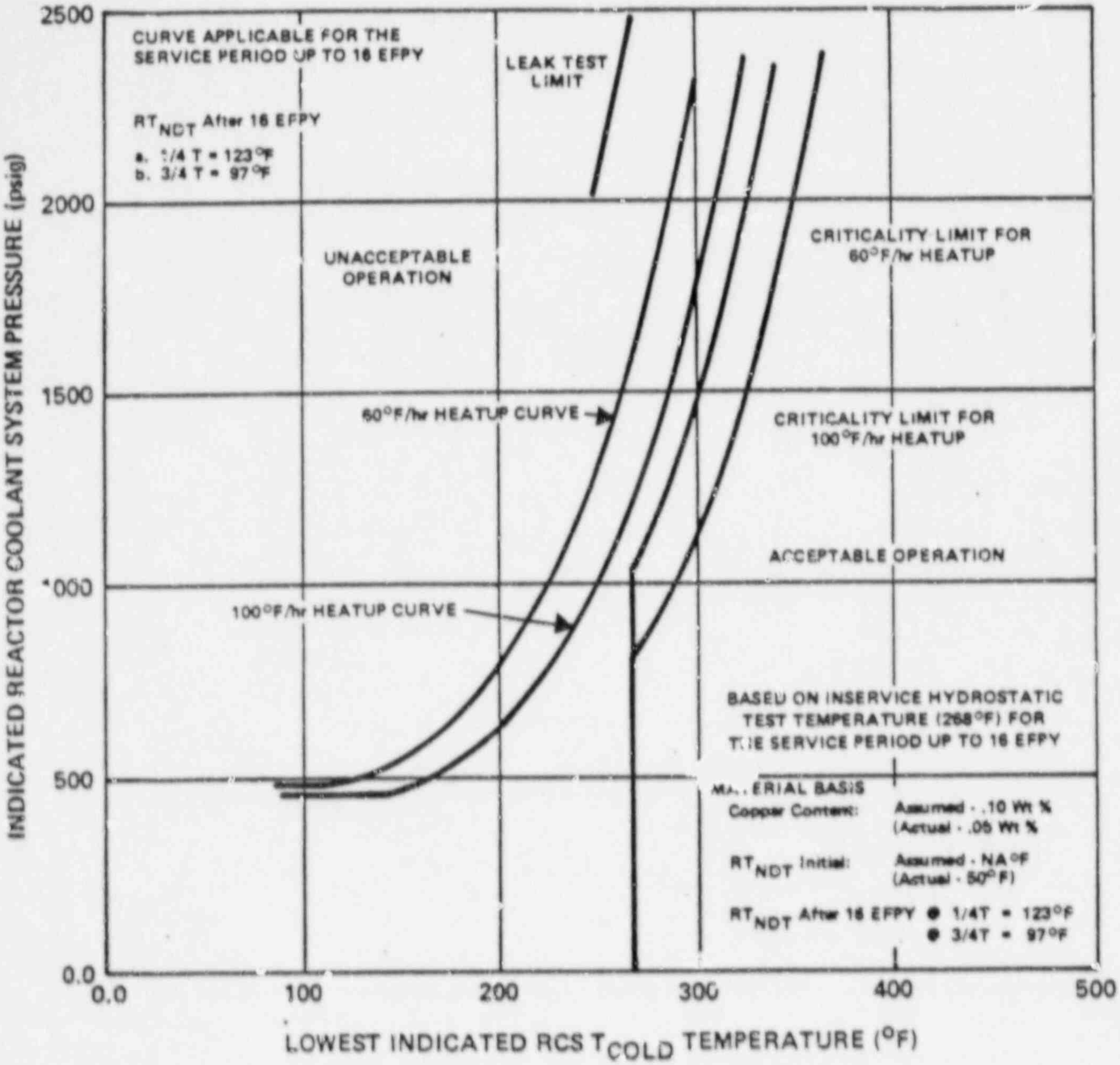


FIGURE 3.4-2b

UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 16 EFPY

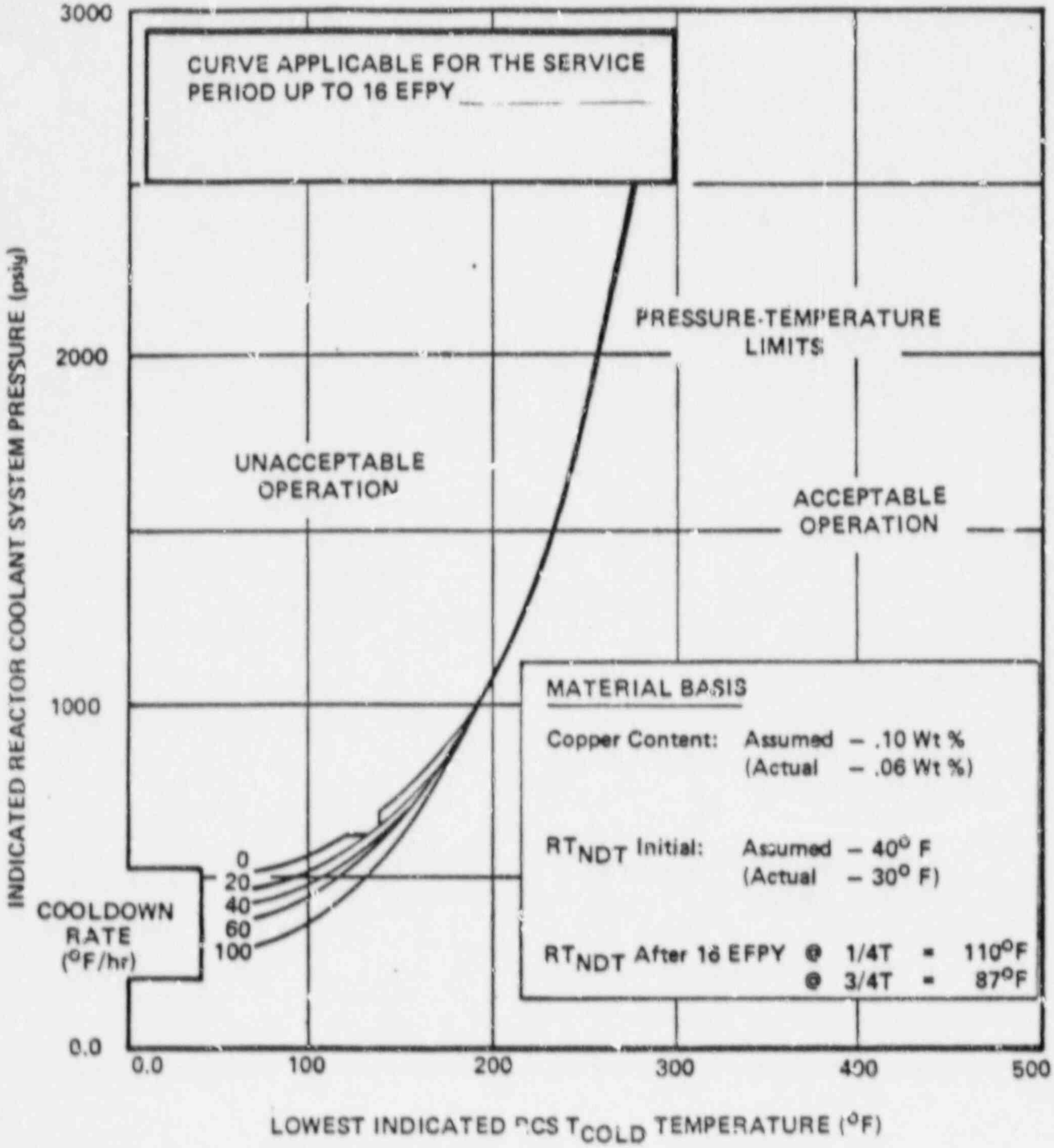


FIGURE 3.4-3a

UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 16 EFPY

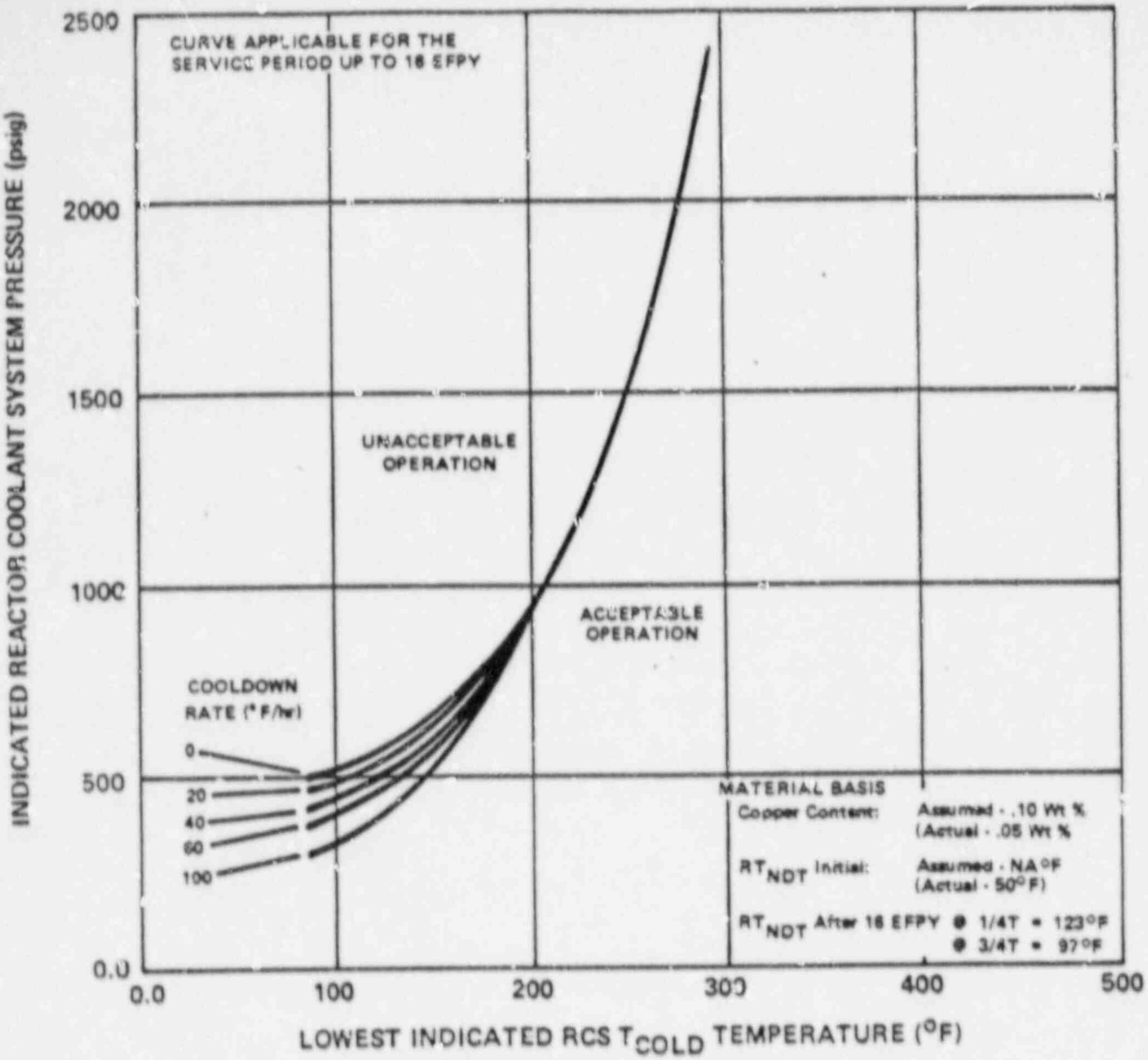


FIGURE 3.4-3b

UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 16 EFPY

REACTOR COOLANT SYSTEM

COLD OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Cold Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with lift settings which vary with RCS temperature and which do not exceed the limits established in ~~Figure 3.4-4, or~~

See insert AI.

b. Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig ± 3%, or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent capable of relieving at least 670 GPM water flow at 470 psig.

APPLICABILITY: MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS as specified in Specification 3.4.9.3.c, above, within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS as specified in Specification 3.4.9.3.c, above, within 8 hours.

c. In the event either the PORVs, the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHR suction relief valves or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

INSERT AI

Figure 3.4-4a (Unit 1), Figure 3.4-4b (Unit 2), or

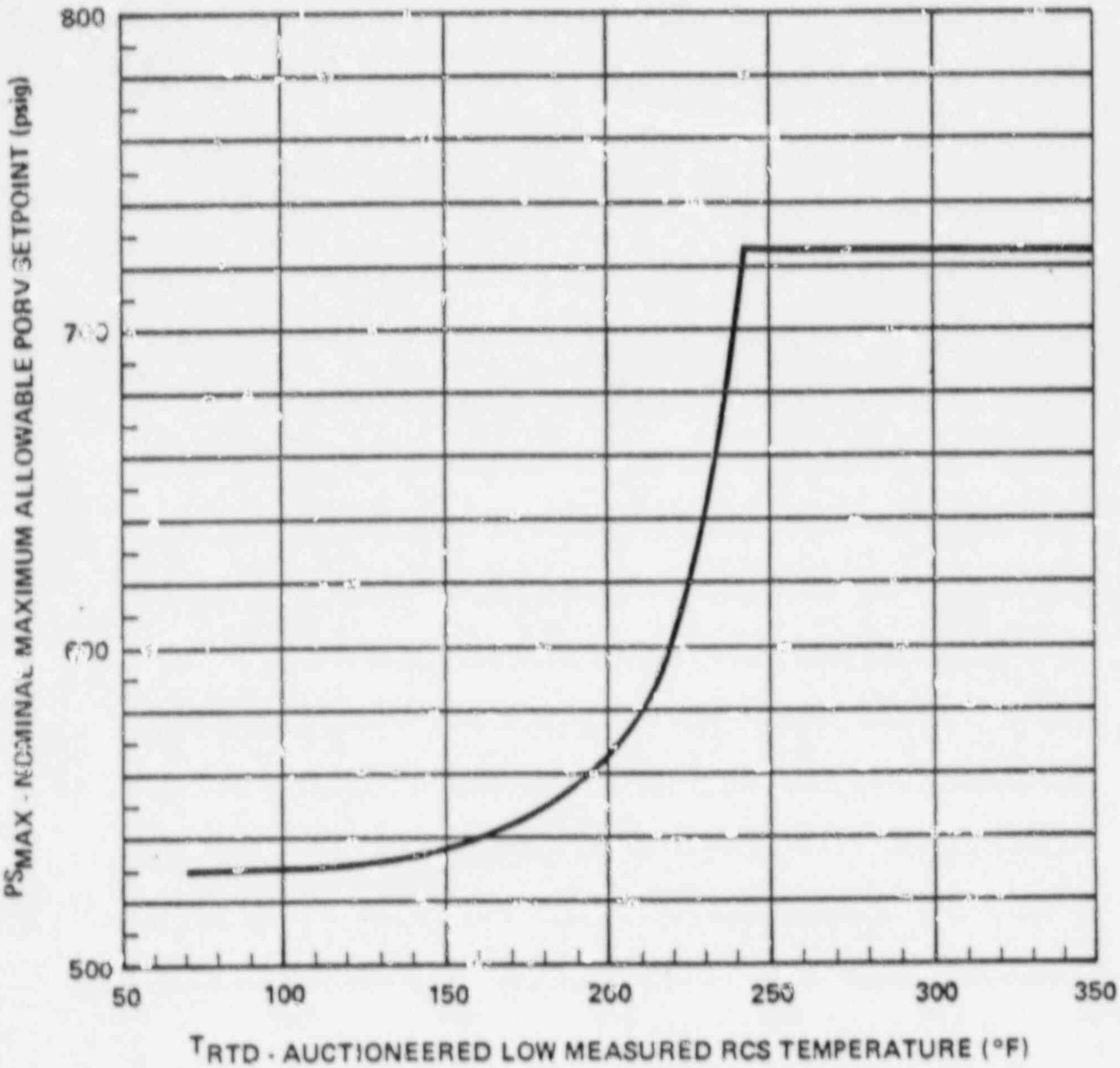


FIGURE 3.4-4a

UNIT 1 MAXIMUM ALLOWABLE NOMINAL PORV SETPOINT
FOR THE COLD OVERPRESSURE PROTECTION SYSTEM

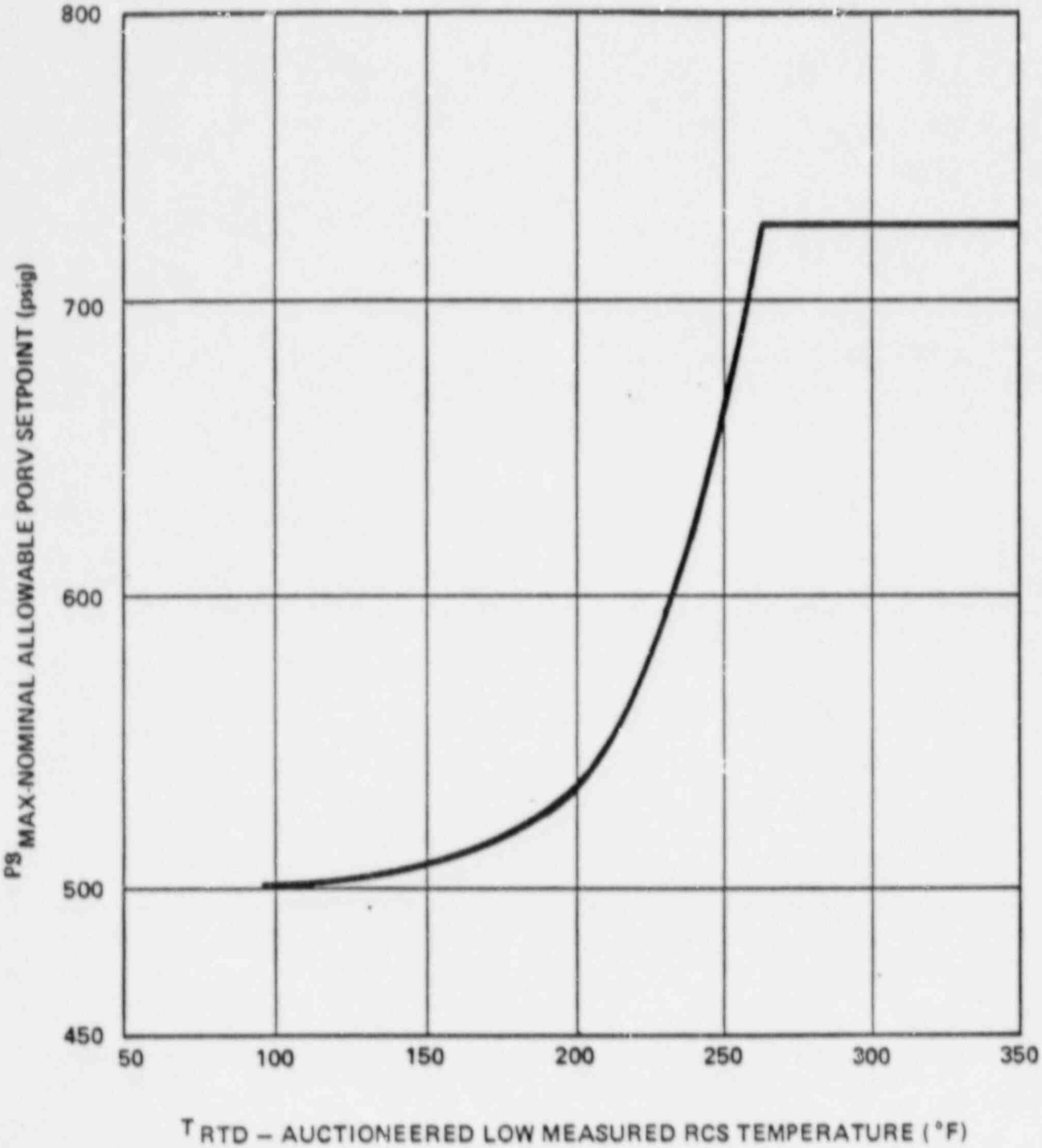


FIGURE 3.4-4b

UNIT 2 MAXIMUM ALLOWABLE NOMINAL PORV SETPOINT FOR THE COLD OVERPRESSURE PROTECTION SYSTEM

(2)

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open,

b. A contained borated water volume of between 6616 (36% of instrument span) and 6854 (64% of instrument span) gallons (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

See insert AJ.

c. A boron concentration of between 1900 and ~~2100 ppm, and~~

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B).

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open. (HV-8808A, B, C, D).

*Pressurizer pressure above 1000 psig.

INSERT AJ

c. A boron concentration of between:

Unit 1 - 1900 ppm and 2600 ppm
Unit 2 - 1900 ppm and 2100 ppm, and

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power lockout switches in the lockout position:

Valve Number	Valve Function	Valve Position
HV-8835	SI Pump Cold Leg. Inj.	OPEN
HV-8840	RHR Pump Hot Leg. Inj.	CLOSED
HV-8813	SI Pump Mini. Flow Isol.	OPEN
HV-8806	SI Pump Suction from RWST	OPEN
HV-8802A, B	SI Pump Hot Leg Inj.	CLOSED
HV-8809A, B	RHR Pump Cold Leg Inj.	OPEN *

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the Containment Emergency Sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 377 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the Containment Emergency Sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

* Either valve may be realigned in MODE 3 for testing pursuant to Specification 4.4.6.2.2.

BORON INJECTION SYSTEM

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

See insert AK.

~~b. A boron concentration of between 2000 ppm and 2100 ppm of boron,~~

c. A minimum solution temperature of 54°F, and

d. A maximum solution temperature of 116°F (TI-10982).

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

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

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed..

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 50°F.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

INSERT AK

b. A boron concentration of between:

Unit 1 - 2400 ppm and 2600 ppm
Unit 2 - 2000 ppm and 2100 ppm

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 L_a the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the test by verifying that the absolute value of the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 45 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks, and

2) Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2;

g. The provisions of Specification 4.0.2 are not applicable.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature (TE-2563, TE-2612, TE-2613) shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location	Tag Numbers*
a. Level 2	TE-2563
b. Level B	TE-2613
c. Level C	TE-2612

*Or local sample at corresponding location

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

place Unit 1

a. With more than one Unit 1 tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the Unit 1 containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In addition, within 90 days of completion of the Unit 1 evaluation, perform an engineering evaluation of Unit 2 containment and provide a Special Report to the Commission in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

place Unit 2

b. With abnormal degradation of the Unit 1 structural integrity as defined in 4.6.1.6.1.1 items b or e, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of Unit 1 containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In addition, within 90 days of completion of the Unit 1 evaluation, perform an engineering evaluation of the Unit 2 containment and provide a Special Report to the Commission in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

place Unit 1

place Unit 2

above

that defined in

c. With any abnormal degradation of the structural integrity other than ACTION a and b at a level below the acceptance criteria of Specification 4.6.1.6, restore the applicable containment to the required level of integrity within 72 hours and perform an engineering evaluation of the applicable containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

place the applicable unit

SURVEILLANCE REQUIREMENTS

4.6.1.6.1.1 Containment Tendons (Unit 1). The structural integrity of the Unit 1 containment tendons shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The structural integrity of the tendons shall be demonstrated by:

a. Determining that a random but representative sample of at least 13 tendons (4 inverted U and 9 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve (HV-2626A&B, HV-2627A&B, HV-2628A&B, HV-2629A&B) shall be OPERABLE and:

a. Each 24-inch containment purge supply and exhaust isolation valve shall be closed and sealed closed, and

b. The 14-inch containment purge supply and exhaust isolation valve(s) shall be closed to the maximum extent practicable but may be open for purge system operation for pressure control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve(s) to be open.

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

ACTION:

a. With a 24-inch containment purge supply and/or exhaust isolation valve open or not sealed closed, close and seal that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 14-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in Specification 3.6.1.7b above, close the open 14-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 24-inch containment purge supply and exhaust isolation valve (HV-2626A, HV-2627A, HV-2628A, HV-2629A) shall be verified to be sealed closed at least once per 31 days.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 At least once per 3 months the containment purge valves with resilient material seals in each sealed closed containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured penetration leakage rate is less than 0.06 La when pressurized to Pa.

4.6.1.7.3 Each 14-inch containment purge supply and exhaust isolation valve (HV-2626B, HV-2627B, HV-2628B, HV-2629B) shall be verified to be closed or open in accordance with Specification 3.6.1.7b at least once per 31 days.

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 shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four 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 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					LIFT SETTING (± 1%)*	ORIFICE SIZE
	SG-1	SG-2	SG-3	SG-4		
1.	PSV 3001	PSV 3011	PSV 3021	PSV 3031	1185 psig	16.0 in^2
2.	PSV 3002	PSV 3012	PSV 3022	PSV 3032	1200 psig	16.0 in^2
3.	PSV 3003	PSV 3013	PSV 3023	PSV 3033	1210 psig	16.0 in^2
4.	PSV 3004	PSV 3014	PSV 3024	PSV 3034	1220 psig	16.0 in^2
5.	PSV 3005	PSV 3015	PSV 3025	PSV 3035	1235 psig	16.0 in^2

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

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination*	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 14 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Emergency Filtration Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 during movement of irradiated fuel or movement of loads over irradiated fuel.

ACTION:

MODES 1, 2, 3 or 4:

With one Control Room Emergency 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.

MODES 5, and 6 during movement of irradiated fuel or movement of loads over irradiated fuel:

a. With one Control Room Emergency 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 Filtration System in the emergency mode.

b. With both Control Room Emergency Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Filtration System, required to be in the emergency mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving movement of irradiated fuel or movement of loads over irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Room Emergency Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to ~~80°F~~ 85°F

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow (FI-12191, FI-12192) through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heater control circuit energized.

SURVEILLANCE REQUIREMENTS (Continued)

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 ~~880~~ 1500 cfm relative to adjacent areas during system operation;

4) Verifying that the heaters dissipate 118 ± 6 kW when tested in accordance with Section 14 of ANSI N510-1980; and

5) Verifying that on a Control Room/Toxic Gas Isolation test signal, the control room isolation dampers close within 6 seconds and the system automatically switches into an isolation mode of operation with flow through the HEPA filters and charcoal adsorbers.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 19,000 cfm ± 10%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal absorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 19,000 cfm ± 10%.

3/4.7.7 PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Piping Penetration Area Filtration and Exhaust Systems shall be OPERABLE.

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

ACTION:

With one Piping Penetration Area Filtration and Exhaust 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.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Piping Penetration Area Filtration and Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow (FI-12629, FI-12542) through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heater control circuit energized.

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:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria of greater than or equal to 99.95% filter retention while operating the system at a flow rate of 15,500 cfm ± 10% (Unit 1) and cfm ± 10% (Unit 2) performing the following tests:

(a) A visual inspection of the piping penetration area filtration and exhaust system shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

(b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

(c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

*The provisions of this specification are not applicable to Unit 2 until its initial entry into MODE 2.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period* **
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

**The provisions of Specification 4.0.2 are not applicable.

e. Functional Tests (Continued)

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.8f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

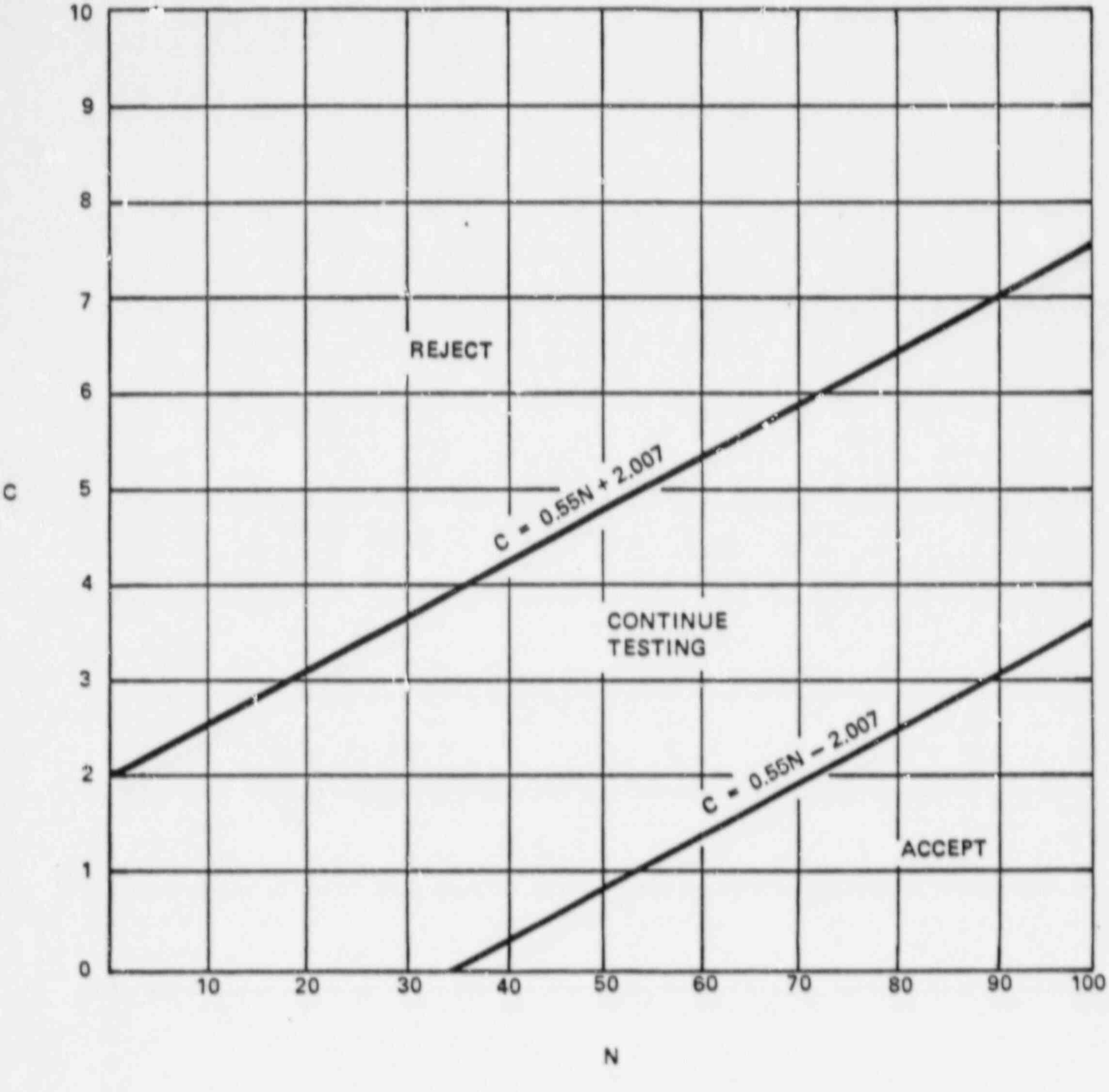


FIGURE 4.7-1

SAMPLE PLAN 2 FOR SNUBBER FUNCTIONAL TEST

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TABLE 3.7-3

AREA TEMPERATURE MONITORING

BUILDING	ROOM NO	MAXIMUM NORMAL TEMP	MAXIMUM ABNORMAL TEMP
FUEL	B006	104	120
AUXILIARY	110	100	104
AUXILIARY	202	100	107
AUXILIARY	203	100	107
AUXILIARY	A017	100	112
AUXILIARY	A047	100	107
AUXILIARY	B017	100	100
AUXILIARY	C113	100	109
AUXILIARY	C120	100	106
AUXILIARY	D053	100	110
AUXILIARY	D068	100	106
AUXILIARY	D072	100	105
AUXILIARY	D075	100	106
AUXILIARY	D119	100	105
AUXILIARY	D121	100	103
CONTROL	147	80	87
CONTROL	149	80	87
CONTROL	A054	100	114
CONTROL	A062	100	103
CONTROL	B065	100	106
CONTROL	B074	100	107
CONTROL	B078	100	104

[Rooms associated with Unit 2 will be added later]

3/4.7.13 DIESEL GENERATOR BUILDING AND AUXILIARY FEEDWATER PUMPHOUSE ESF HVAC SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.13 The diesel generator building and auxiliary feedwater pumphouse ESF HVAC systems shall be OPERABLE with:

a. At least two ESF supply fans and associated dampers per diesel generator train, and

b. At least three ESF auxiliary feedwater pumphouse HVAC systems.

APPLICABILITY: Whenever the associated diesel generator or auxiliary feedwater pumps are required to be operable.

ACTION:

a. With 1 or more supply fans to a given diesel generator train inoperable, restore the inoperable fan(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 1 or more ESF auxiliary feedwater pumphouse HVAC system inoperable, follow the ACTION specified in specification 3.7.1.2 for an inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.13 The diesel generator building and auxiliary feedwater pumphouse ESF HVAC system shall be demonstrated OPERABLE:

a. At least once per 18 months by:

   1) Verifying that the diesel generator ESF supply fans, 1/2-1566-B7-001 (train A) and 1/2-1566-B7-002 (train B) start automatically and the associated intake and discharge dampers actuate to their correct position on their train associated diesel generator running signal.

   2) Verifying that diesel generator ESF supply fans 1/2-1566-B7-003 and 1/2-1566-B7-004 start automatically and the associated intake and discharge dampers actuate to their correct position on a high diesel generator building temperature signal coincident with diesel generator running signal.

b. At least once per 18 months by:

   1) Verifying that the auxiliary feedwater pumphouse ESF supply fans, 1/2-1593-B7-001 and 1/2-1593-B7-002 and associated shutoff dampers actuate to their correct position on a high room temperature signal.

SURVEILLANCE REQUIREMENTS

4.7.13 (Continued)

2) Verifying that the ESF outside air intake and exhaust dampers for the turbine driven auxiliary feedwater pump actuate to the correct position on a turbine driven auxiliary feedwater pump automatic start signal.

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

1) A day tank containing a minimum volume of 650 gallons (52% of instrument span) of fuel (LI-9018, LI-9019),

2) A separate Fuel Storage System containing a minimum volume of 68,000 gallons of fuel (76% of instrument span) (LI-9024, LI-9025), and

3) A separate fuel transfer pump.

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

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either diesel generator has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for each such diesel generator, separately, within 24 hours unless the diesel generator is already operating. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With either diesel generator inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours*#. Restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

#The diesel shall not be rendered inoperable by activities performed to support testing pursuant to the Action Statement (e.g., an air roll).

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and, if the diesel generator became inoperable due to any cause other than preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours*, unless the OPERABLE diesel generator is already operating#. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of 3.8.1.1, Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel generator OPERABILITY per Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under the Action Statement for an OPERABLE diesel generator or a restored to OPERABLE diesel generator satisfies the diesel generator test requirement of Action Statement a or b.

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d. With one diesel generator inoperable in addition to ACTION b. or c. above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and
2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators separately by performing the requirements of Specification 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours#, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow Action Statement a with the time requirement of that Action Statement based

Superscript

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#The diesel shall not be rendered inoperable by activities performed to support testing pursuant to the Action Statement (e.g., an air roll).

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with supplier's certification;

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

f. At least once per 92 days and from new fuel prior to addition to the storage tank obtain a sample and verify that the neutralization number is less than 0.2 and the mercaptan content is less than 0.01% ~~#, ##~~. Superscript

g. At least once per 184 days by:

1) Verifying the diesel starts* from ambient conditions and the generator voltage and frequency are 4160 + 170, -410 volts and 60 ± 1.2 Hz within 11.4 seconds after the start signal. The diesel generator shall be started for this test by using one of the signals listed in Surveillance Requirement 4.8.1.1.2.a.4. This test, if it is performed so it concides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

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*All engine starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress on the diesel engine.

#Mercaptan content shall not be required to be verified within specification for new fuel prior to its addition, for up to 15,000 gallons of fuel added to the tank, if the last tank sample had a mercaptan content of less than 0.007%. All subsequent new fuel addition will require mercaptan content verification prior to its addition until the tank contents are verified to be less than 0.007%.

~~##Neutralization number will not have to be verified less than 0.2, for new fuel prior to its addition, until 60 days after license issuance. Until that time verification of new fuel specifications will be completed within 30 days of addition.~~

Specification

Synchronized

2) Verifying the generator is synchonized, loaded to an indicated value of 6100 - 7000 kW*** in less than or equal to 60 seconds, and operates with a load of 6800-7000 kW*** for at least 60 minutes This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

h. At least once per 18 months,** during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturers' recommendations for this class of standby service;

2) Verifying the diesel generator capability to reject a load of greater than or equal to 671 kW (motor-driven auxiliary feedwater pump) while maintaining voltage at 4160 + 240, -410 volts and speed of less than 484 rpm (less than nominal speed plus 75% of the difference between nominal speed and the Overspeed Trip Setpoint); and recovering voltage to within 4160 + 170, -410 volts within 3 seconds.

3) Verifying the diesel generator capability to reject a load of 7000 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11.5 seconds,* energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 +170, -410 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 +170, -410 volts and 60 ± 1.2 Hz within 11.4 seconds after the

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*All engine starts for the purpose of surveillance testing as required by Specification 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

**For any start of a diesel, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate this test.

auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11.5 seconds,* energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 +170, -410 volts and 60 ± 1.2 Hz during this test; and

c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure, high jacket water temperatures and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 7600 to 7700 kW,** and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6800-7000 kW.** The generator voltage and frequency shall be 4160 + 170, - 410 volts and 60 ± 1.2 Hz within 11.4 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.# Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2h.6)b);##

8) Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 7000 kW;

9) Verifying the diesel generator's capability to:

Superscripts

Specification

*All engines starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate the test.

#Failure to maintain voltage and frequency requirements due to grid disturbances does not render a 24-hour test as a failure.

##If Specification 4.8.1.1.2h.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at the load required by Surveillance Requirement 4.8.1.1.2.a5 kW for 1 hour or until operating temperature has stabilized.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	≤ 4	Once per 31 days
≥ 2**	≥ 5	Once per 7 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul, including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 and four tests in accordance with the 184-day testing requirement of Surveillance Requirement 4.8.1.1.2.f. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

D.C. SOURCES

1/2

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one train related pair of 125-V battery banks (either 125-V battery banks 1AD1B and 1CD1B or 125-V battery banks 1BD1B and 1DD1B) and one associated full-capacity charger per required battery bank shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery bank and/or both chargers inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery bank and at least one charger to OPERABLE status as soon as possible, and within 8 hours, provide relief capabil[illegible] the Reactor Coolant System in accordance with Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

SAFETY-RELATED MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection bypass devices of each safety-related motor-operated valve given in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection bypass devices for any one or more safety-related motor-operated valves inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection bypass devices shall be verified to be OPERABLE.

a. Following maintenance on the valve motor starter, and

b. Following any periodic testing during which the thermal overload device was temporarily placed in force.

c. At least once per 18 months, during shutdown.

TABLE 3.8-1

SAFETY-RELATED
MOTOR OPERATED VALVES THERMAL OVERLOAD
PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION
1/2HV-1668A	Nuclear Service Clg. Twr. A Return
1/2HV-1668B	Nuclear Service Clg. Twr. A Fans Bypass
1/2HV-1669A	Nuclear Service Clg. Twr. B Return
1/2HV-1669B	Nuclear Service Clg. Twr. B Fans Bypass
1/2HV-1806	CTMT Air Cool A7001/A7002 CW Inlet
1/2HV-1807	CTMT Air Cool A7003/A7004 CW Inlet
1/2HV-1808	CTMT Air Cool A7005/A7006 CW Inlet
1/2HV-1809	CTMT Air Cool A7007/A7008 CW Inlet
1/2HV-1822	CTMT Air Cool A7001/A7002 CW Outlet
1/2HV-1823	CTMT Air Cool A7003/A7004 CW Outlet
1/2HV-1830	CTMT Air Clr A7005/A7006 CW Outlet
1/2HV-1831	CTMT Air Clr A7007/A7008 CW Outlet
1/2HV-1974	Aux Comp CW Trn B Return Iso
1/2HV-1975	Aux Comp CW Trn A Return Iso
1/2HV-1978	Aux Comp CW Trn B Supply Iso
1/2HV-1979	Aux Comp CW Trn A Supply Iso
1/2HV-2041	Reactor Coolant Pumps Thermal Barrier ACCWS Outlet Header
1/2HV-2134	Reactor Cavity Clg Coil E7001 Inlet Iso
1/2HV-2135	Reactor Cavity Clg Coil E7002 Inlet Iso
1/2HV-2138	Reactor Cavity Clg Coil E7001 Outlet Iso
1/2HV-2139	Reactor Cavity Clg Coil E7002 Outlet Iso
1/2HV-12114	CR Outside Air Intake Iso
1/2HV-12115	CR Outside Air Intake Iso
1/2HV-12118	CB CR Filter Units N7001 Inlets
1/2HV-12119	CB CR Filter Units N7002 Inlets
1/2HV-12128	CB CR Filter Units N7001 Outlets
1/2HV-12129	CB CR Filter Units N7002 Outlets
1/2HV-12130	CB CR Return Air Fans B7005 Inlets
1/2HV-12131	CB CR Return Air Fans B7006 Inlets
1/2HV-12727	CB SR Battery Rm Exh B7002 Damper
1/2HV-12742	CB SF Battery Rm Exh B7001 Damper
1/2HV-12748	CB SF Battery Rm Exh B7003 Damper
1/2HV-12749	CB SF Battery Rm Exh B7004 Damper
1/2HV-19051	Thermal Barrier Cooling Wtr RCP 001
1/2HV-19053	Thermal Barrier Cooling Wtr RCP 002
1/2HV-19055	Thermal Barrier Cooling Wtr RCP 003
1/2HV-19057	Thermal Barrier Cooling Wtr RCP 004
1/2HV-8920	Safety-Injection Pump Miniflow Isolation
1/2HV-2624A	CTB Post LOCA Purge Exhaust Iso
1/2HV-2624B	CTB Post LOCA Purge Exhaust Iso
1/2HV-2626A	CTMT Bldg Norm Purge Supply Iso

TABLE 3.8-1 (Continued)

SAFETY-RELATED MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION
1/2HV-2627A	CTB Norm Purge Supply Iso
1/2HV-2628A	CTMT Bldg Norm Purge Exhaust Iso
1/2HV-2629A	CTB Norm Purge Exhaust Iso
1/2HV-5106	Aux FDW Pump Turbine
1/2HV-5113	Conds Stor TK V4002 to Pump P4001
1/2HV-5118	Conds Stor TK V4002 to Pump P4002
1/2HV-5119	Conds Stor TK V4002 to Pump P4003
1/2HV-5120	Aux FDW Pump P4001 Discharge Trn C
1/2HV-5122	Aux FDW Pump P4001 Discharge Trn C
1/2HV-5125	Aux FDW Pump P4001 Discharge Trn C
1/2HV-5127	Aux FDW Pump P4001 Discharge Trn C
1/2HV-5132	Aux FDW Pump P4002 Discharge Trn B
1/2HV-5134	Aux FDW Pump P4002 Discharge Trn B
1/2HV-5137	Aux FDW Pump P4003 Discharge Trn A
1/2HV-5139	Aux FDW Pump P4003 Discharge Trn A
1/2FV-5154	Aux FDW Pump P4002 Miniflow
1/2FV-5155	Aux FDW Pump P4003 Miniflow
1/2HV-8438	Charging Pump B Discharge
1/2HV-8471A	Alt Charging Pump A Suction
1/2HV-8471B	Alt Charging Pump B Suction
1/2HV-8485A	Charging Pump A Discharge
1/2HV-8485B	Charging Pump B Discharge
1/2HV-8508A, B	Charging Pump Miniflow Iso to RWST
1/2HV-8509A, B	Charging Pump Miniflow Iso to RWST
1/2HV-9380A	CTMT ATM Unit 1 SVCE Air
1/2HV-9380B	CTMT ATM Unit 1 SVCE Air
1/2HV-12005	Trn B Aux FDW Pump Rm Inlet Damper
1/2HV-12006	Trn A Aux FDW Pump Rm Inlet Damper
1/2HV-12050	DGB Exh Fan B7001 Disch Damper (Trn A)
1/2HV-12051	DGB Exh Fan B7003 Disch Damper (Trn A)
1/2HV-12052	DGB Exh Fan B7005 Disch Damper (Trn A)
1/2HV-12053	DGB Exh Fan B7002 Disch Damper (Trn B)
1/2HV-12054	DGB Exh Fan B7004 Disch Damper (Trn B)
1/2HV-12055	DGB Exh Fan B7006 Disch Damper (Trn B)
1/2HV-8000A, B	PORV Blockline
1/2HV-8147	Reg. Hx Tube Outlet to RCS Alternate Chg
1/2HV-8146	Reg. Hx Tube Outlet to RCS Normal Chg
1/2HV-8100, 8112	No 1 Seal Leakoff
1/2HV-8103A, B, C, D	RCP No 1 Seal from Chg
1/2HV-8111A, B, 8110	Chg Pump Miniflow
1/2LV-0112C, B	VCT Discharge Header
1/2LV-0112E, D	SIS RWST Discharge to Chg/SI Pump Suction
1/2HV-8104	CVCS Boric Acid Filler to Charging Pump Suction

TABLE 3.8-1 (Continued)

SAFETY-RELATED MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION
1/2HV-8105; 8106	Chg Pump to RCS Isolation
1/2HV-8807A, B; 8924	HHSI Suction to Chg/SI Suction
1/2HV-8801A, B	BIT Discharge
1/2HV-8808A, B, C, D	Accumulator Discharge
1/2HV-8811A, B	Containment Emergency Sump Isolation
1/2HV-8812A, B	RHR Suction from RWST
1/2HV-8809A, B	RHR Discharge Header
1/2HV-8804A	RHR Hx No. 1 Outlet to Charge Pump
1/2HV-8804B	RHR Hx No. 2 Outlet to SI Pumps
1/2HV-8806	RWST Discharge Header to SI Pumps
1/2HV-8923A, B	SI Pump Suction Isolation
1/2HV-8813; 8814	SI Pump Miniflow
1/2HV-8821A, B	SI Pump Crosschannel
1/2HV-8835	SI Pump Discharge to Cold Legs
1/2HV-8840	RHR Pump Discharge to Hot Legs
1/2HV-8802A, B	SI Pump Discharge Header
1/2HV-8701A, B; 8702A, B	RHR Suction from RCS Hot Legs 1, 4
1/2FV-0610; 0611	RHR Miniflow
1/2HV-8716A, B	RHR Cross Connect
1/2HV-9002A, B	Spray Pump Containment Emergency Sump Isolation
1/2HV-9003A, B	Spray Pump Containment Emergency Sump Isolation
1/2HV-9017A, B	Spray Pump Suction from RWST
1/2HV-9001A, B	Spray Pump Discharge Header
1/2HV-8994A, B	Spray Additive Tank Discharge
1/2HV-11600	NSCW Pump Discharge Isolation
1/2HV-11605	NSCW Pump Discharge Isolation
1/2HV-11606	NSCW Pump Discharge Isolation
1/2HV-11607	NSCW Pump Discharge Isolation
1/2HV-11612	NSCW Pump Discharge Isolation
1/2HV-11613	NSCW Pump Discharge Isolation
1/2PV-2550A	Piping Penetration Room to Atmosphere
1/2PV-2551A	Piping Penetration Room to Atmosphere
1/2HV-3009	TDAFP Steam Supply Isolation
1/2HV-3019	TDAFP Steam Supply Isolation
1/2HV-8116	Charging Pump Discharge Boron Injection
1/2PV-15129	TDAFP Trip and Throttle Valve
1/2HV-2582A	CTB Cooling Unit A7001
1/2HV-2582B	CTB Cooling Unit A7002
1/2HV-2583A	CTB Cooling Unit A7003
1/2HV-2583B	CTB Cooling Unit A7004
1/2HV-2584A	CTB Cooling Unit A7005
1/2HV-2584B	CTB Cooling Unit A7006
1/2HV-2585A	CTB Cooling Unit A7007
1/2HV-2585B	CTB Cooling Unit A7008

REFUELING OPERATIONS

3/4.9.12 FUEL HANDLING BUILDING POST ACCIDENT VENTILATION SYSTEM (COMMON SYSTEM)

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Post Accident Ventilation Systems shall be OPERABLE.

either

APPLICABILITY: Whenever irradiated fuel is in ~~the~~ storage pool.

ACTION:

either either

a. With one Fuel Handling Building Post Accident Ventilation System inoperable, fuel movement within ~~the~~ storage pool or crane operation with loads over ~~the~~ storage pool may proceed provided the OPERABLE Fuel Handling Building Post Accident Ventilation 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.

b. With no Fuel Handling Building Post Accident Ventilation System OPERABLE, suspend all operations involving movement of fuel within either ~~the~~ storage pool or crane operation with loads over ~~the~~ storage pool until at least one Fuel Handling Building Post Accident Ventilation System is restored to OPERABLE status.

either

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Post Accident Ventilation Systems shall be demonstrated OPERABLE:

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 heater control circuit energized;

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:

(13)

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microcurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (µCi/ml)
1. Batch Waste Release Tanks(2)	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	$5x10^{-7}$
a. Waste-Monitor Tank 1901-T6-009			I-131	$1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
b. Waste-Monitor Tank 1901-T6-010				
	P Each Batch	M Composite(4)	H-3	$1x10^{-5}$
c. Drainage of Systems			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite(4)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	1x10-6
2. Continuous Releases(5)	Continuous(6)	W Composite(6)	Principal Gamma Emitters(3)	$5x10^{-7}$
			I-131	1x1 $^{-6}$
a. Waste Water retention basin	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Continuous(6)	M Composite(6)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continous(6)	Q Composite(6)	Sr-89, Sr-90	$1x10^{-8}$
			Fe-55	$1x10^{-6}$

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}); and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(2)A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

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

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE NOTATIONS

* 3b may be ommitted provided the absence of a primary to secondary leak has been demonstrated; that is, the gamma activity in the secondary water does nc exceed background by more than 20%.

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(2)	Thirty-six routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each meteorological sector in the 6 mile range from the site; and The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations Three samples from close to the three SITE BOUNDARY locations, in different sectors; One sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q; and One sample from a control location, as for example a population center 10 to 20 miles distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;(3) and gamma isotopic analysis(4) of composite (by location) quarterly.
3. Waterborne			
a. Surface(5)	One sample upstream One sample downstream	Composite sample over 1-month period.(6)	Gamma isotopic analysis(4) monthly. Composite for tritium analysis quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)			
b. Drinking	Two samples at each of one to three of the nearest water treatment plants that could be affected by its discharge. Two samples at a control location.	Composite sample of river water near intake at each water treatment plant over 2-week period(6) when I-131 analysis is performed, monthly composite otherwise; and grab sample of finished water at each water treatment plant every 2 weeks or monthly, as appropriate.	I-131 analysis on each sample when the dose calculated for the consumption of the water is greater than 1 mrem per year(7). Composite for gross beta and gamma isotopic analyses(4) monthly. Composite for tritium analysis quarterly.
c. Sediment from Shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis(4) semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 3 miles distance having the highest dose potential. If there are none, then one sample from milking animals(8) in each of three areas between 3 and 5 miles distance where doses are calculated to be greater than 1 mrem per yr.(7)	Semimonthly.	Gamma isotopic(4,9) analysis semimonthly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
a. Milk	One sample from milking animals[8] at a control location about 10 miles distant or beyond and preferably in a wind direction of lower prevalence.		
b. Fish	At least one sample of any commercially and recreationally important species in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis[4] on edible portions.
	At least one sample of any species in areas not influenced by plant discharge.		
	At least one sample of any anadromous species in vicinity of plant discharge.	During spring spawning season.	Gamma isotopic analyses[4] on edible portion.
c. Grass or Leafy Vegetation	One sample from two onsite locations near the SITE BOUNDARY in different sectors.	Monthly during growing season.	Gamma isotopic[4,9]
	One sample from a control location at about 15 miles distance.	Monthly during growing season.	Gamma isotopic[4,9]

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	GRASS OR LEAFY VEGETATION (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-95	400				
Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			200	
La-140	100			300	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[1] [2]

LOWER LIMIT OF DETECTION (LLD)[3]

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	GRASS OR LEAFY VEGETATION (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			
Co-60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	60			60		
La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.
**If no drinking water pathway exists, a value of 15 pCi/l may be used.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute com ance with the terms of an individual Limiting Condition for Operation and associate ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence wou violate the intent of a specification. For example, Specification 3.5.2 requires t independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirement one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, i both the required ECCS subsystems are inoperable, within 1 hour measures must be in ated to place the unit in at least HOT STANDBY within the next 6 hours, and in at l HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specifi- tion 3.0.3, if both the required Containment Spray Systems are inoperable, within measures must be initiated to place the unit in at least HOT STANDBY within the nex 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOW within the subsequent 24 hours. It is acceptable to initiate and complete a reduct in OPERATIONAL MODES in a shorter time interval than required in the ACTION stateme and to add the unused portion of this allowable out-of-service time to that provide for operation in subsequent lower OPERATION MODE(S). Stated allowable out-of-servi times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperabili is discovered but the times provided for achieving a mode reduction are not applica if the inoperability is discovered in a mode lower than the applicable mode. For e ple if the Containment Spray System was discovered to be inoperable while in STARTU the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would s be available before the plant would be required to be in COLD SHUTDOWN However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provi to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or othe specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statemen

The intent of this provision is to ensure that facility operation is not initi with either required equipment or systems inoperable or other specified limits bein exceeded.

178, 182 gallons (Unit 1), 87,720 gallons (Unit 2).

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions as defined by Specification 3/4.1.1.1 (MODES 1 and 2) and Specification 3/4.1.1.2 (MODES 3 and 4) 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 31740 gallons usable volume of 7000 ppm borated water from the boric acid storage tanks or ~~87720 gallons~~ usable volume of ~~2000 ppm~~ borated water from the refueling water storage tank (RWST).

2400 ppm (Unit 1), 2000 ppm (Unit 2)

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.

2400 ppm (Unit 1), 2000 ppm (Unit 2)

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 (MODE 5) after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4570 gallons usable volume of 7000 ppm borated water from the boric acid storage tanks or ~~12630 gallons~~ usable volume of ~~2000 ppm~~ borated water from the RWST.

41,202 gallons (Unit 1), 12,630 gallons (Unit 2)

The contained water volume limits provided in Specifications 3/4.1.2.5 and 3/4.1.2.6 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 8.5 and 10.5 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 OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section are necessary to ensure that the following requirements are met at all times during normal operation. By observing that the RCCAs are positioned above their respective insertion limits during normal operation,

1. At any time in life for MODE 1 and 2 operation, the minimum SHUTDOWN MARGIN will be maintained. For operational MODES 3, 4, 5, and 6, the reactivity condition consistent with other specifications will be maintained with all RCCAs fully inserted by observing that the boron concentration is always greater than an appropriate minimum value.

2. During normal operation the enthalpy rise hot channel factor, $F^N_{\Delta H}$, will be maintained within acceptable limits.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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.

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 two 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.

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

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^{N}_{\Delta H}$

The limits on heat flux hot channel factor 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 ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 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;

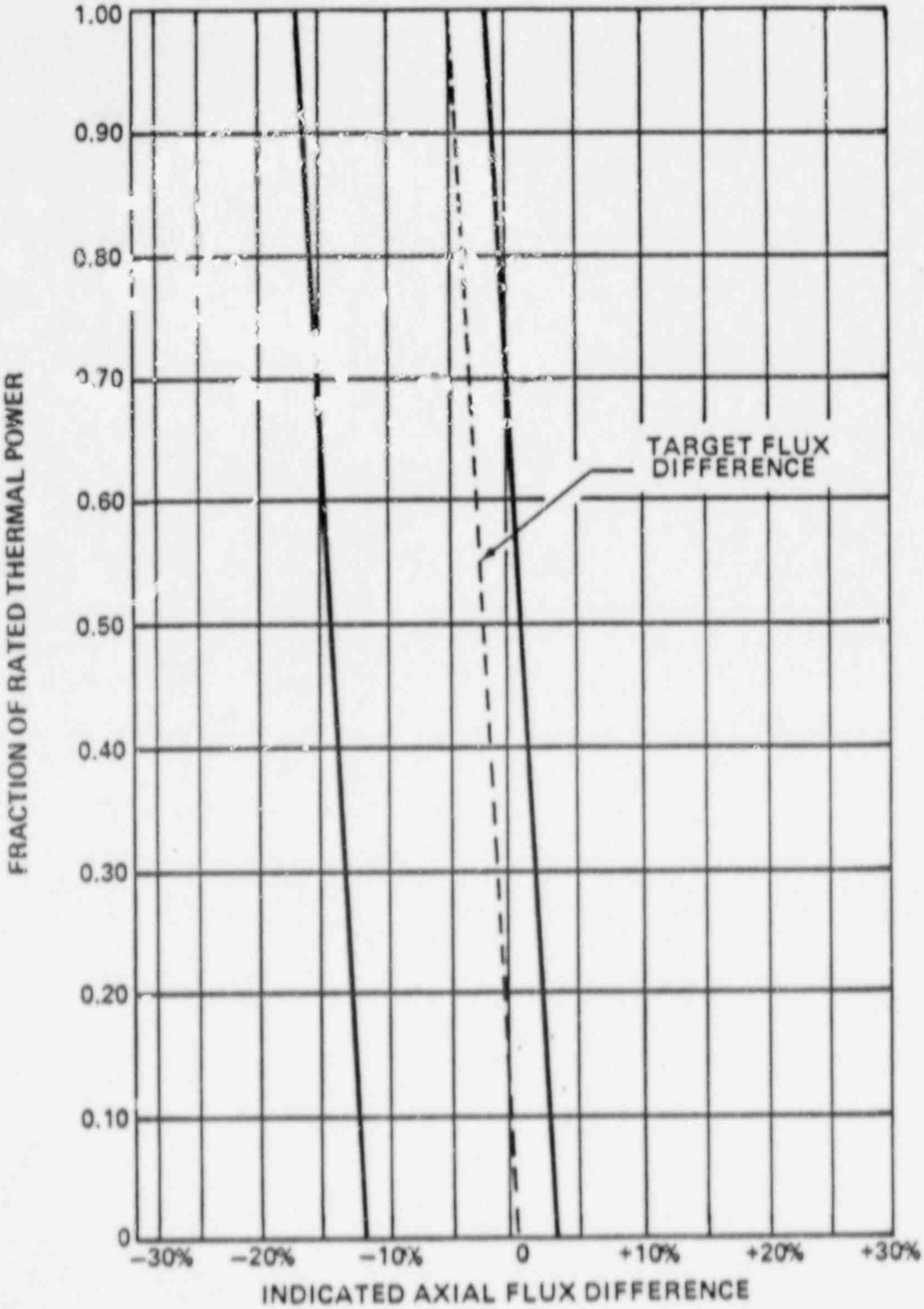

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

(7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment fan coolers start and automatic valves position, (11) Nuclear Service Cooling and Component Cooling water pumps start and automatic valves position, and (12) Control Room Ventilation Emergency Actuation Systems start.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 With pressurizer pressure below the P-11 setpoint, allows manual block of safety injection actuation on low pressurizer pressure signal. Allows manual block of safety injection actuation and steam line isolation on low compensated steam line pressure signal and allows steam line isolation on high steam line negative pressure rate. With pressurizer pressure above the P-11 setpoint, defeats manual block of safety injection actuation on low pressurizer pressure and safety injection and steam line isolation on low steam line pressure and defeats steam line isolation on high steam line negative pressure rate.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, initiates a turbine trip, and inhibits feedwater control valve modulation.

See insert AL.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

INSERT AL

The Source Range High Flux at Shutdown Alarm Setpoint is an analysis assumption for mitigation of a Boron Dilution Event in MODES 3, 4, and 5.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation on Unit 1 is shared with Unit 2 and the seismic instrumentation and corresponding Technical Specifications meet the ~~requirements~~ recommendations of Regulatory Guide 1.12, Revision 1, April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the specific count should be made for gases (i.e., xenons and kryptons) and particulates (i.e., cobalt and cesiums) in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

The identification of 95% of the gross specific activity by definition does not obligate VEGP into calculating $\bar{E}$ every time gross activity is determined.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS ~~[To Be Revised To Reflect Unit 2]~~

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with ~~Figures 3.4-2 and 3.4-3~~ for the service period specified thereon:

See insert AM.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures ~~3.4-2 and 3.4-3~~ define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

INSERT AM

Figures 3.4-2a and 3.4-3a (Unit 1), Figures 3.4-2b and 3.4-3b (Unit 2)

~~[To Be Revised To Reflect Unit 2]~~

PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The auxiliary spray shall not be used if the temperature difference between the pressurizer and the auxiliary spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Replace with insert AN.

~~The heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 are applicable to Vogtle-Unit 1 for up to 16 EFPY. The most limiting material for Vogtle-Unit 1 has an initial RT_{NDT} of 30°F and a copper content of 0.06 WT %, whereas the heatup and cooldown curves in Figures 3.4-2 and 3.4-3 are based on an initial RT_{NDT} of 40°F and a copper content of 0.10 WT %.~~

Heatup and cooldown limit curves are calculated using the most limiting value ~~of the nil-ductility reference temperature~~, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown for Units 1 and 2 in Table B 3/4.4-1 a and b, respectively. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of

INSERT AN

The heatup and cooldown limit curves shown in Figures 3.4-2a and 3.4-3a are applicable to Unit 1 for up to 16 EFPY and are based on Westinghouse-developed generic curves which were developed assuming a 40°F initial RT_{NDT} and a copper content of 0.10 WT% for the most limiting material. These curves are applicable to Unit 1 since its most limiting material (Table B 3/4.4-1a) has both a lower initial RT_{NDT} (30°F) and a lower copper content (0.05 WT%). These curves, however, are not applicable to Unit 2, since its most limiting material (Table B 3/4.4-1b) has a higher initial RT_{NDT} (50 compared to 40°F). Separate heatup and cooldown limit curves were developed based on the actual material properties of the most limiting material for Unit 2 up to 16 EFPY. The Unit 2 curves are shown in Figures 3.4-2b and 3.4-3b.

TABLE B 3/4.4-1a

Unit 1 REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU (%)	NI (%)	P (%)	T_{NDT} ~~NDTT~~ (°F)	50 FT-LB 35 MIL TEMP (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF ENERGY ~~NMWD*~~ (FT-LB)	MWD** (FT-LB)
Closure Head Dome	B8807-1	A533BCL.1	.16	.67	.006	-50	75	15	88	-
Closure Head Torus	B8808-1	A533BCL.1	.14	.56	.010	-30	68	8	85	-
Closure Head Flange	B8801-1	A508CL.2	-	.70	.011	20	<40	20	132	-
Vessel Flange	B8802-1	A508CL.2	-	.71	.014	0	<60	0	119	-
Inlet Nozzle	B8809-1	A508CL.2	-	.86	.011	-20	<10	-20	107	-
Inlet Nozzle	B8809-2	A508CL.2	-	.84	.014	-10	<50	-10	95	-
Inlet Nozzle	B8809-3	A508CL.2	-	.82	.013	-10	<10	-10	117	-
Inlet Nozzle	B8809-4	A508CL.2	-	.87	.014	-20	<10	-20	105	-
Outlet Nozzle	B8810-1	A508CL.2	-	.82	.006	-10	<50	-10	>124	-
Outlet Nozzle	B8810-2	A508CL.2	-	.79	.006	-10	<50	-10	>100	-
Outlet Nozzle	B8810-3	A508CL.2	-	.77	.006	-10	<50	-10	>102	-
Outlet Nozzle	B8810-4	A508CL.2	-	.80	.006	-10	<10	-10	>75	-
Nozzle Shell	B8804-1	A533BCL.1	.14	.62	.011	-10	88	28	94	-
Nozzle Shell	B8804-2	A533BCL.1	.10	.58	.006	-40	75	15	104	-
Nozzle Shell	B8804-3	A533BCL.1	.14	.69	.013	-30	100	40	92	-
Inter. Shell	B8805-1	A533BCL.1	.08	.59	.004	0	60	0	90	-
Inter. Shell	B8805-2	A533BCL.1	.08	.59	.004	-10	80	20	100	-
Inter. Shell	B8805-3	A533BCL.1	.06	.60	.003	-20	90	30	107	-
Lower Shell	B8606-1	A533BCL.1	.05	.59	.005	-50	80	20	116	-
Lower Shell	B8606-2	A533BCL.1	.05	.58	.009	-10	80	20	113	-
Lower Shell	B8606-3	A533BCL.1	.06	.64	.007	-20	70	10	118	-
Bottom Head Torus	B8813-1	A533BCL.1	.13	.50	.009	-40	50	-10	88	-
Bottom Head Dome	B8812-1	A533BCL.1	.10	.53	.009	-40	32	-28	122	-
Inter & Lower Shell Vertical Weld Seams and Girth Seam	G1.43	SAW	.03	.10	.007	-80	<-20	-80	>129	-

[To Be Revised To Reflect Unit 2]

~~*Normal to major working directions~~

~~**Major working direction~~ Limiting Material.

[Upper Shelf energy,

TABLE B 3/4.4-1b

UNIT 2 REACTOR VESSEL TOUGHNESS

Component	Comp. Code	ASME Material Type	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD* (ft-lb)
Closure head dome	R9-1	A533B Cl. 1	0.07	0.61	0.008	-40	-30	1[illegible]3
Closure head torus	R10-1	A533B Cl. 1	0.07	0.64	0.010	-30	0	84
Closure head flange	R7-1	A508 Cl. 2	-	0.72	0.011	10	10	130
Vessel flange	R1-1	A508 Cl. 2	-	0.87	0.011	-60	-60	115
Inlet nozzle	B9806-1	A508 Cl. 2	0.07	0.84	0.010	-50	-50	119
Inlet nozzle	B9806-2	A508 Cl. 2	0.06	0.83	0.009	-40	-40	128
Inlet nozzle	R5-1	A508 Cl. 2	0.09	0.87	0.008	-20	-20	147
Inlet nozzle	R5-2	A508 Cl. 2	0.08	0.85	0.009	-20	-20	134
Outlet nozzle	R6-3	A508 Cl. 2	-	0.69	0.011	-10	-10	122
Outlet nozzle	R6-4	A508 Cl. 2	-	0.66	0.010	-10	-10	140
Outlet nozzle	B9807-3	A508 Cl. 2	-	0.66	0.005	-30	-30	116
Outlet nozzle	B9807-4	A508 Cl. 2	-	0.66	0.010	10	10	132
Nozzle shell	R3-1	A533B Cl. 1	0.20	0.67	0.015	0	20	79
Nozzle shell	R3-2	A533B Cl. 1	0.20	0.67	0.015	0	40	79
Nozzle shell	R3-3	A533B Cl. 1	0.15	0.62	0.010	-10	60	84
Intermediate shell	R4-1	A533B Cl. 1	0.06	0.64	0.009	-20	10	95
Intermediate shell	R4-2	A533B Cl. 1	0.05	0.62	0.009	-10	10	104
Intermediate shell	R4-3	A533B Cl. 1	0.05	0.59	0.009	0	30	84
Lower shell	B8825-1	A533B Cl. 1	0.05	0.59	0.006	-20	40	83
Lower shell	R8-1	A533B Cl. 1	0.06	0.62	0.007	-20	40	87
Lower shell	B8628-1	A533B Cl. 1	0.05	0.59	0.007	-20	50	85
Bottom head torus	R12-1	A533B Cl. 1	0.17	0.64	0.012	-20	-20	89
Bottom head dome	R11-1	A533B Cl. 1	0.10	0.62	0.008	-30	-30	115
Intermediate and lower shell vertical weld seams	G1.60	SAW	0.07	0.13	0.007	-10	-10	147
Intermediate to lower shell girth weld seam	E3.[illegible]3	SAW	0.06	0.12	0.007	-50	-30	90

*Upper Shelf Energy; NMWD - normal to major working direction.

**Limiting Material.

PRESSURE/TEMPERATURE LIMITS (Continued) See insert AO.

Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of ~~Figures 3.4-2 and 3.4-3~~ include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 16.3-3 of the VEGP FSAR. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup

INSERT AO

Figures 3.4-2a and 3.4-3a (Unit 1), Figures 3.4-2b and 3.4-3b (Unit 2)

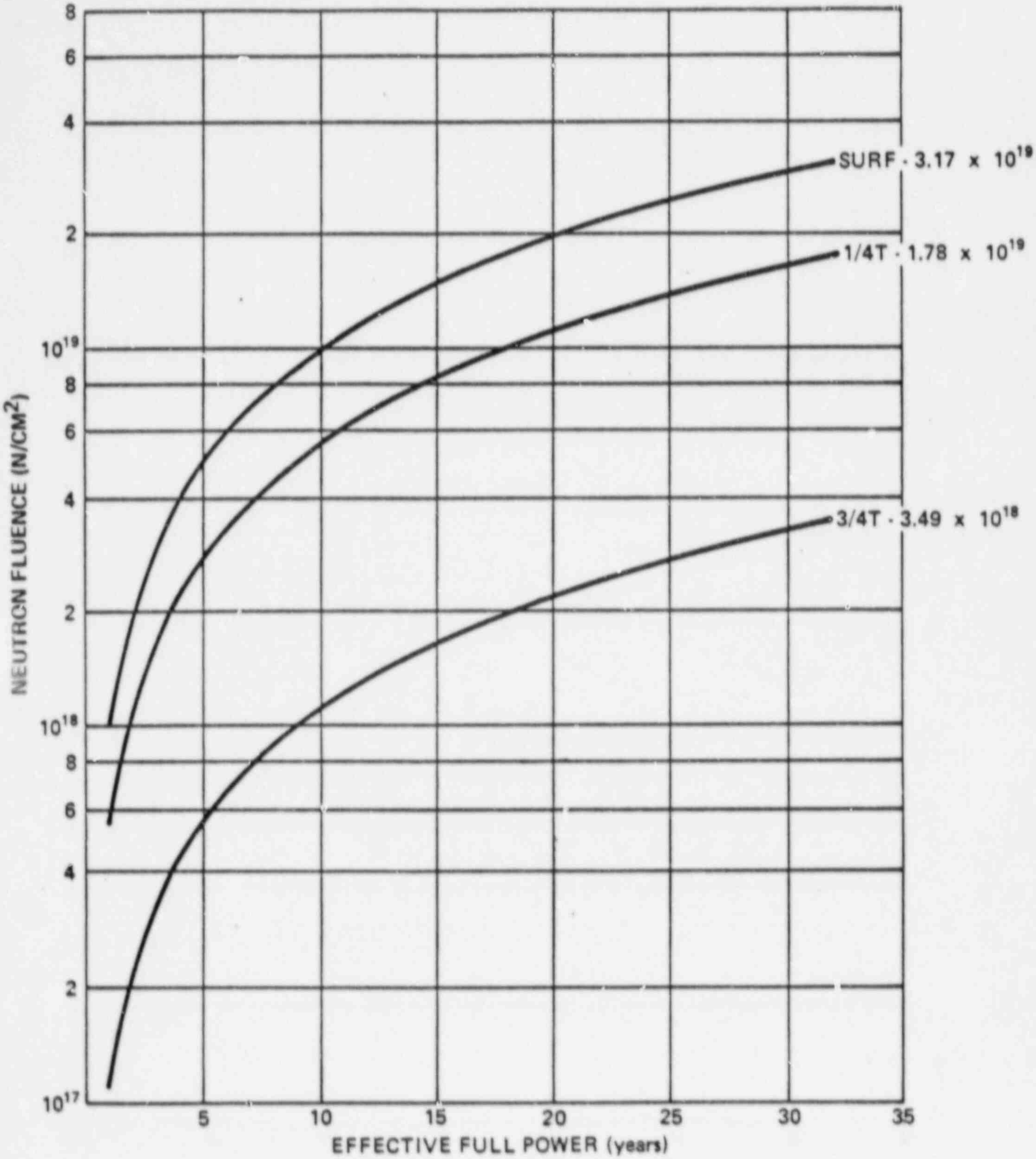

FIGURE B 3/4.4-1

UNITS 1 AND 2 FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

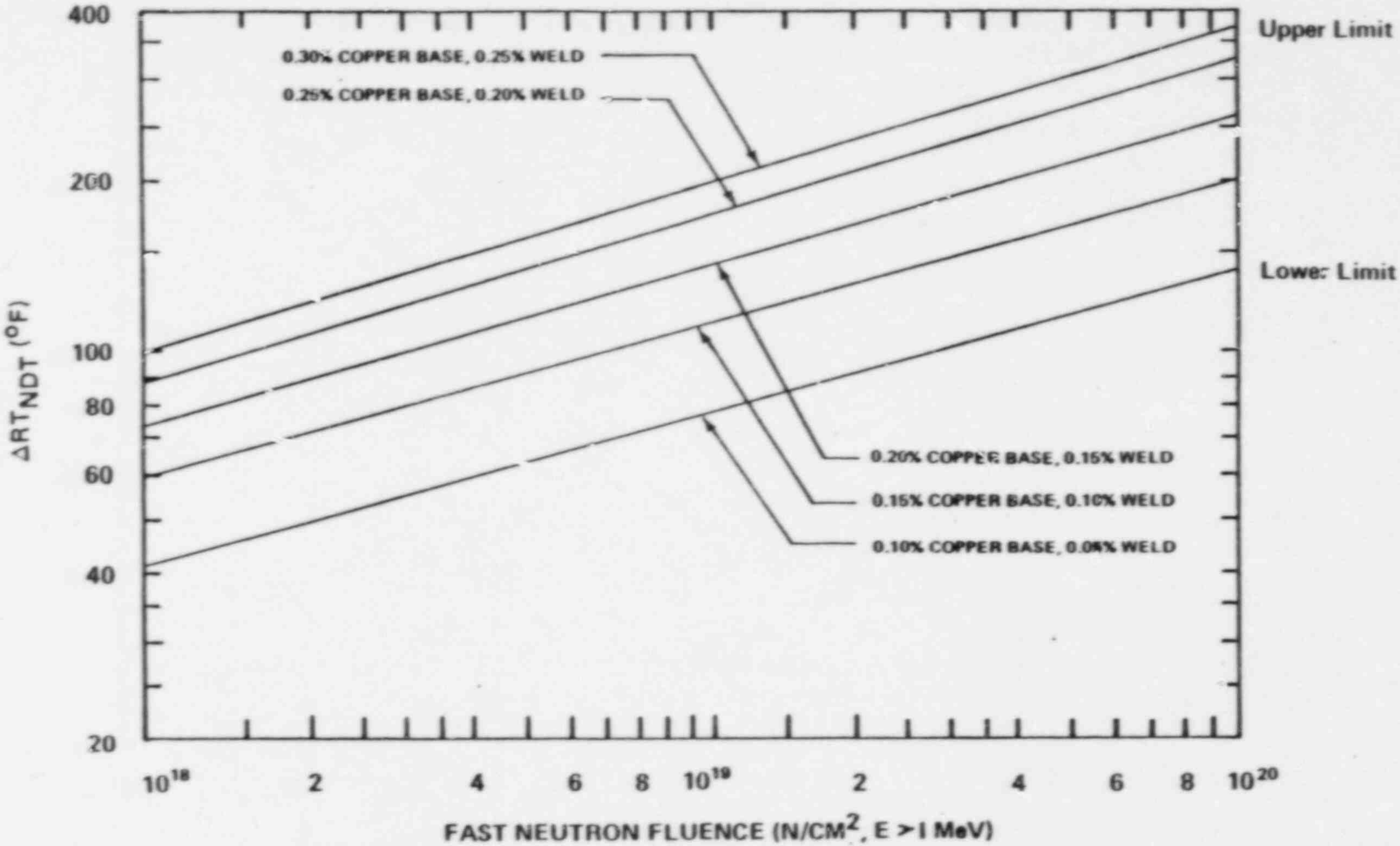

FIGURE B 3/4.4-2

UNITS 1 & 2 EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO RADIATION AT 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Next, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Spaces

Finally, the new 10CFR50 Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This rule states that the minimum metal temperature of the closure flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Westinghouse Plants). For ~~Vogtle~~ Unit 1 the minimum temperature of the closure flange and vessel flange regions is 140°F, since the limiting RT_{NDT} is 20°F (see Table B 3/4-4.1). The Vogtle Unit 1 heatup curve shown on Figure 3-4.2 is not impacted by the new 10CFR50 rule. However, the Vogtle Unit 1 cooldown curve shown in Figure 3-4.3 is impacted by the new 10CFR50 rule.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

See insert AP.

COLD OVERPRESSURE PROTECTION SYSTEMS

The OPERABILITY of two PORVs, two RHR suction relief valves or an RCS vent capable of relieving at least 670 gpm water flow at 470 psig ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of all three charging pumps and subsequent injection into a water-solid RCS.

INSERT AP

For Unit 2, the minimum temperature of the closure flange and vessel flange regions is 130°F, since the limiting RT_{NDT} is 10°F (Table B 3/4.4-1b). The Unit 2 heatup curve shown in Figure 3.4-2b and the cooldown curve shown in Figure 3.4-3b are not impacted by the new 10 CFR 50 rule.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

See insert 4Q.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

INSERT AQ

The minimum boron concentration must ensure the reactor core will remain subcritical during the accumulator injection period of a small-break LOCA.

ECCS SUBSYSTEMS (Continued)

The limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis.

See insert AU.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow ≥ 3.0 FT²) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

8.0 (Unit 1), 8.5 (Unit 2)

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between ~~3.5~~ and 10.5 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.

INSERT AU

The surveillance requirements for leakage testing of ECCS check valves ensure a failure of one valve will not cause an intersystem LOCA. In MODE 3, with either HV-8809A or B closed for ECCS check valve leak testing, adequate ECCS flow for core cooling in the event of a LOCA is assured.

BASES

FOR

SECTION 3/4.6

CONTAINMENT SYSTEMS SPECIFICATIONS

FOR

WESTINGHOUSE

ATMOSPHERIC TYPE CONTAINMENT

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of each the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 41.9 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force and the tensile tests of the tendon strands for Unit 1, and the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment and the Type A leakage test for both units are sufficient to demonstrate this capability. (The tendon strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.) Unit 1 and Unit 2 containments satisfy the recommendations of Regulatory Guide 1.35, Revision 2, Position C.1.3. Therefore, Unit 2 containment is subject to visual inspection only.

The Surveillance Requirements for demonstrating the structural integrity of each containment is in compliance with the recommendations of Revision 2 of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 24-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4. Sealed closed isolation valves are isolation valves under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, the use of blind flanges, or removal of power to the valve operator.

CONTAINMENT VENTILATION SYSTEM (Continued)

The use of the containment purge lines is restricted to the 14-inch purge supply and exhaust isolation valves since, unlike the 24-inch valves, the 14-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, should be used to justify the opening of these isolation valves.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System both provide post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

8.0 (Unit 1), 8.5 (Unit 2

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between ~~8.5~~ and 10.5 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 solution volume limits (3700-4000 gallons) represent the required solution to be delivered (i.e., the delivered solution volume is that volume above the tank discharge) These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

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% (1304 psig) of its design pressure of 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, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 18,607,220 lbs/h which is 123% of the total secondary steam flow of 15,135,453 lbs/h at 100% RATED THERMAL POWER. 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 3.7-1.

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 basis:

For four loop operation

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

Where:

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

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

109 = Power Range Neutron Flux-High Trip Setpoint for four 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,

BASES

3/4.7.13 DIESEL GENERATOR BUILDING AND AUXILIARY FEEDWATER PUMPHOUSE ESF HVAC SYSTEMS

The operation of the diesel generator building and auxiliary feedwater pumphouse ESF HVAC systems ensures that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment served by these systems.

23

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974 and Appendix A to Generic Letter 84-15, "Proposed Staff Position to Improve and Maintain Diesel Generator Reliability." When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based on the recommendations of Regulatory Guides 1.9, Revision 2 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December, 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979, Appendix A to Generic Letter 84-15 and Generic Letter 83-26, "Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests."

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," and 484-1975 "Recommended Practice for Installation Design and Installation of Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.10 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance. A list of containment penetration conductor overcurrent protective devices and feeder breakers to isolation transformers between 480 V class 1E busses and non-class 1E equipment is provided in Table 16.3-5 of the VEGP FSAR.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The bypassing of the motor-operated valves thermal overload protection except during periodic testing ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection continuously are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

(24)

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the Boron Dilution Accident in the safety analysis. ~~The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly,~~ the boron concentration value of 2000 ppm or greater ~~includes a conservative uncertainty allowance of 50 ppm boron.~~

See insert AR.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

3/4.9.5 COMMUNICATIONS

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

INSERT AR

ensures a K_{eff} of 0.95 or less and includes a conservative allowance for calculational uncertainties of 100 ppm of boron.

FIGURE 5.1-1

EXCLUSION AREA

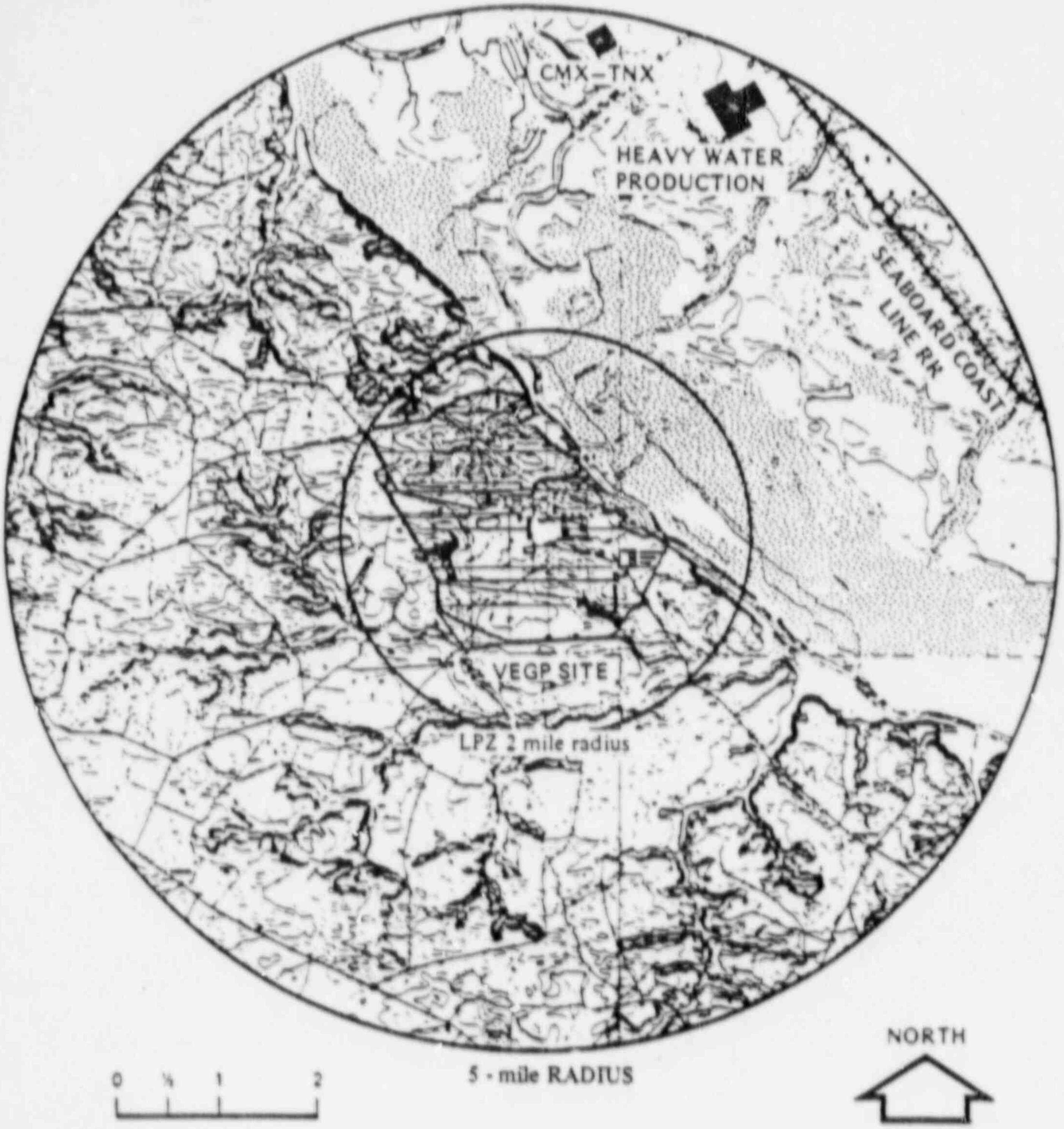

FIGURE 5.1-2

LOW POPULATION ZONE

5.6 FUEL STORAGE

CRITICALITY

Unit 1

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 0.91% Δk/k for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal 10.6 inch center-to-center distance between fuel assemblies placed in the storage racks.

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

2 Unit 2

5.6.1.~~3~~ The ~~West~~ spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of ~~2.4%~~ 1.13% Δk/k for uncertainties as described in Section 4.3 of the FSAR~~, and~~

b. A nominal spacing of 10.58 inches in the North-South direction and 10.4 inches in the East-West direction between fuel assemblies placed in the storage racks.

3

Blank line

5.6.1.~~4~~ The k_{eff} for new fuel for the first core loading stored dry in the ~~West~~ spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 194'-1½".

CAPACITY

5.6.3 The spent fuel storage pools are designed to contain sufficient storage rack locations for long-term storage. Currently, the Unit 1 pool contains two storage racks with a combined capacity of 288 fuel assemblies. The Unit 2 pool may contain up to 20 storage racks with a combined capacity of 2098 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to > 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	> 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	[illegible] cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F and < 625°F.
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > [illegible]-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to ≥ 1481 psig.

6.1 RESPONSIBILITY

6.1.1 The General Manager - ~~Vogtle Nuclear Operations (GMVNO)~~, Nuclear Plant shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The ~~GMVNO~~ General Manager - Nuclear Plant will annually reissue a directive that emphasizes the primary management responsibility of the onshift Operations Supervisor (or during his absence from the control room, the individual designated to assume the command functions) for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

6.2 ORGANIZATION

~~OFFSITE~~ ONSITE AND OFFSITE ORGANIZATIONS

See insert AS

6.2.1 ~~The offsite organization for plant management and technical support shall be as shown in Figure 6.2-1.~~

PLANT STAFF

6.2.2 The plant organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. When fuel is in either reactor at least one operator licensed on the applicable unit shall be in the control room. In addition, while either unit is in MODE 1, 2, 3, or 4 at least one senior operator licensed# on the applicable unit(s) shall be in the control room.

Superscript

c. An individual* who has successfully completed the Initial Technician Training portion of the Health Physics Training Program or its equivalent shall be on site when fuel is in either reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. Administrative procedures shall be developed and implemented to limit the working hours of plant staff in performance of safety-related functions (e.g., licensed Senior Operators, licensed Operators, key Health Physics Technicians, key non-licensed operators, and key maintenance personnel).

#If a single Senior Operator does not hold a Senior Operator's license on both units, two or more Senior Operators who in combination are licensed as Senior Operators on both units may fulfill this requirement.

*This individual may be absent for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

INSERT A5

6.2.1 Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The General Manager - Nuclear Plant shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President - Nuclear shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

PLANT STAFF (Continued)

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. (This work week may consist of 12-hour shift schedules.) However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the applicable department superintendent, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual excess overtime shall be reviewed monthly by the General Manager - ~~Vogtle Nuclear Operations~~, or his designee to assure that excessive hours were authorized and that they do not become routine.

Nuclear Plant

(DELETED)

FIGURE 6.2-1

OFFSITE ORGANIZATION

(DELETED)

FIGURE 6.2-2

PLANT ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

TWO UNITS WITH A COMMON CONTROL ROOM

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
OS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
NLO	3*	3*	3*
STA	1***	none	1***

OS - Operations Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
NLO - Non-Licensed Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Operations Supervisor from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Operations Supervisor from the control room while either unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

*At least one of the required individuals must be assigned to the designated position for each unit.

**At least one licensed Senior Operator or Licensed Senior Operator Limited to Fuel Handling who has not other concurrent responsibilities must be present during CORE ALTERATIONS on either unit.

***The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Operations Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as stated in the Policy Statement on Engineering Expertise on Shift, dated October 28, 1985.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, which may indicate areas for improving plant safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant safety to the ~~Senior~~ Vice President-Nuclear. ~~Operations through the Manager Nuclear Performance and Radiological Safety.~~

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Senior Vice President - Nuclear. ~~Operations through the Manager Nuclear Performance and Radiological Safety.~~

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the plant for transients and accidents, and in plant design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 TRAINING

6.3.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Plant Training and Emergency Preparedness Manager. Personnel will meet the minimum education and experience recommendations of ANSI N18.1-1971 and, for licensed staff, ~~Appendix A of~~ 10 CFR ~~55.~~ 55.59 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, before they are considered qualified to perform all duties independently. Prior to meeting the recommendations of ANSI N18.1-1971, personnel may be trained to perform specific tasks and will be qualified to perform those tasks independently.

See insert AT.

*Not responsible for sign-off function.

INSERT AT

Personnel who complete an accredited program which has been endorsed by the NRC shall meet the requirements of the accredited program in lieu of the above.

6.4 REVIEW AND AUDIT

6.4.1 PLANT REVIEW BOARD (PRB)

FUNCTION

General Manager - Nuclear Plant

6.4.1.1 The PRB shall function to advise the ~~GMVNO~~ on all matters related to nuclear safety.

COMPOSITION

6.4.1.2 The PRB shall be composed of Department Superintendents or Managers, or supervisory personnel reporting directly to Department Superintendents or Managers from the departments listed below:

Operations
Maintenance
Quality Control
Health Physics
Nuclear Safety and Compliance
Engineering Support

A senior health physicist is acceptable for the Health Physics Department PRB representative. The chairman, his alternate and other members and their alternates of the PRB shall be designated by the ~~GMVNO~~.

General Manager - Nuclear Plant

ALTERNATES

6.4.1.3 No more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.4.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman or his designated alternate.

QUORUM

6.4.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.4.1.6 The PRB shall be responsible for:

a. Review of 1) procedures which establish plant-wide administrative controls to implement the QA program or Technical Specifications surveillance program, 2) procedures for changing plant operating modes, 3) emergency and abnormal operating procedures, 4) procedures for effluent releases of radiological consequences, and 5) fuel handling procedures.

RESPONSIBILITIES (Continued)

b. Review of 1) programs required by Specification 6.7.4 and changes thereto, and 2) proposed procedures and changes to procedures which involve an unreviewed safety question as per 10 CFR 50.59.

c. Review of all proposed tests and experiments that affect nuclear safety;

d. Review of all proposed changes to the Technical Specifications;

e. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety, including proposed changes to Chapter 16.3 of the Vogtle Final Safety Analysis Report (FSAR);

f. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Senior Vice President-Nuclear ~~Operations~~ and to the Safety Review Board;

g. Review of all REPORTABLE EVENTS;

h. Review of plant operations to detect potential hazards to nuclear safety;

i. Performance of special reviews, investigations, or analyses and reports thereon as requested by the ~~GMVNO~~ or the Safety Review Board;

j. Review of the Security Plan and implementing procedures and submittal of recommended changes to the ~~GMVNO~~ and the Safety Review Board;

General Manager-Nuclear Plant

k. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the ~~GMVNO~~ and the Safety Review Board;

l. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the ~~Senior~~ Vice President Nuclear ~~Operations~~ and to the Safety Review Board;

m. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems; and

n. Review of the Fire Protection Program and Implementing procedures and submittal of recommended changes to the ~~GMVNO~~.

6.4.1.7 The PRB shall:

a. Recommend in writing to the ~~GMVNO~~ approval or disapproval of items considered under Specification 6.4.1.6a. through e. prior to their implementation;

RESPONSIBILITIES (Continued)

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.4.1.6a. through f. constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Senior Vice President-Nuclear ~~Operations~~ and the Safety Review Board of disagreement between the PRB and the ~~GMNO~~; however, the ~~GMNO~~ shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

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RECORDS

6.4.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the ~~Senior~~ Vice President-Nuclear ~~Operations~~ and the Safety Review Board.

6.4.2 SAFETY REVIEW BOARD (SRB)

FUNCTION

6.4.2.1 The SRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The SRB shall report to and advise the ~~Senior~~ Vice President-Nuclear ~~Operations~~ on those areas of responsibility specified in Specifications 6.4.2.7 and 6.4.2.8.

COMPOSITION

6.4.2.2 The SRB shall be organized as one board for all GPC Nuclear power plants. The SRB shall be composed of a minimum of five persons who, as a group, provide the expertise to review and audit the operation of a nuclear power plant. The chairman and other members shall be appointed by the ~~Senior~~ Vice President-Nuclear ~~Operations~~ or other such person as he may designate. The composition of the SRB shall meet the requirements of ANSI N18.7-1976.

AUDITS (continued)

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months;

k. The Emergency Plan and implementing procedures (at least once per 12 months);

l. The Security Plan and implementing procedures (at least once per 12 months).

RECORDS

6.4.2.9 Records of SRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SRB meeting shall be ~~prepared~~, approved, and forwarded to the ~~Senior~~ Vice President-Nuclear ~~Operations~~ within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.4.2.7 shall be prepared, approved, and forwarded to the ~~Senior~~ Vice President-Nuclear ~~Operations~~ within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.4.2.8 shall be forwarded to the ~~Senior Executive Vice President~~, Senior Vice President-Nuclear ~~Operations and to the management~~ positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5 REPORTABLE EVENT ACTION

6.5.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.72 and Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, ~~and the~~ results of this review ~~shall be~~ submitted to the SRB and the ~~Senior~~ Vice President-Nuclear ~~Operations~~.

6.6 SAFETY LIMIT VIOLATION

6.6.1 The following actions shall be taken in the event a Safety Limit is violated:

a. In accordance with 10 CFR 50.72, the NRC Operations Center shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The ~~Senior~~ Vice President-Nuclear ~~Operations~~, the SRB, PRB, and the ~~GMVNO~~ shall be notified within 24 hours.

General Manager - Nuclear Plant

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The Licensee Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, ~~and to the~~ PRB, SRB, the ~~GMVNO~~ and the ~~Senior~~ Vice President-Nuclear ~~Operations~~ within 30 days after discovery of the event.

d. Critical operation of the affected unit shall not be resumed until authorized by the Nuclear Regulatory Commission.

6.7 PROCEDURES AND PROGRAMS

6.7.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance for effluent and environmental monitoring;

h. Fire Protection Program Implementation; and

i. Technical Specifications Improvement Program implementation. (FSAR Chapter 16.3)

6.7.2 Each procedure of 6.7.1 above, and changes thereto, shall be reviewed as set forth in administrative procedures and approved by either the ~~GMVNO~~ or the department head of the responsible department prior to implementation with the exception of the following which shall be approved by the ~~GMVNO~~:

General Manager - Nuclear Plant

PROCEDURES AND PROGRAMS (Continued)

1) procedures which establish plant-wide administrative controls (which implement the quality assurance program and the Technical Specifications surveillance program),

2) unit operating procedures (UOPs)

3) emergency operating procedures (EOPs)

4) abnormal operating procedures (AOPs)

5) procedures for implementing the security plan, emergency plan, and the fire protection program, and

6) fuel handling procedures.

PRB responsibilities for procedures are delineated in 6.4.1.

6.7.3 Temporary changes to procedures of Specification 6.7.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license; and

c. The change is documented, reviewed in accordance with Specification 6.7.2 and approved by the ~~GMVNO~~ or department head of the responsible department within 14 days of implementation.
General Manager - Nuclear Plant

6.7.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the following:

1) Residual Heat Removal System
2) Containment Spray System (excluding NaOH Subsystem)
3) Safety Injection (excluding Boron Injection & Accumulators)
4) Chemical and Volume Control System (Letdown, Boron Recycle, and Charging Pumps)
5) Post Accident Processing System
6) Gaseous Waste Processing System
7) Nuclear Sampling System (Pressurizer steam and liquid sample lines, Reactor Coolant sample lines, RHR sample lines, CVCS Demineralizer and Letdown Heat Exchanger sample lines only)

9 10

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.8.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be established for at least each reload core and shall be maintained available in the Control Room. The limits shall be established and implemented on a time scale consistent with normal procedural changes.

The analytical methods used to generate the F_{xy} limits shall be those previously reviewed and approved by the NRC*. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the F_{xy} limits for all core planes containing Bank "D" control rods and all unrodded core planes along with the plot of predicted $F_Q^T.Pr_{e}l$ vs axial core height (with the limit envelope for comparison) shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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*WCAP 8385 "Power Distribution Control and Load Following Procedures" and WCAP 9272.A "Westinghouse Reload Safety Evaluation Methodology."

6.12 PROCESS CONTROL PROGRAM (PCP) (Continued)

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon approval by the ~~GMVNO~~. General Manager - Nuclear Plant

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The ODCM shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon approval by the ~~GMVNO~~. General Manager - Nuclear Plant

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS (Continued)

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PRB.

b. Shall become effective upon approval by the ~~GMVNO~~.

General Manager - Nuclear Plant