TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

For Unit 2

(i) for $q_t - q_b$ between -52% and +5.5%, $f_1(\Delta q) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER,

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -52%, the N-16 Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +5.5%, the N-16 Trip Setpoint shall be automatically reduced by 2.17% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.8% of span for Unit 1 or ~~2.67%~~ of span for Unit 2.

2.88%

PROOF & REVIEW

1

PROOF & REVIEW

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

2

The DNB design basis is that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence level that DNB will not occur when the minimum DNBR is at the DNBR limit. In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters and fuel fabrication parameters are considered such that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

b

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis limit value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

PROOF & REVIEW

SAFETY LIMITS

BASES

REACTOR CORE (continued)

5 These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, and a reference for e axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} \; [1.0 + PF_{\Delta H} \; (1.0 - P)]$$

where: P = the fraction of RATED THERMAL POWER (RTP),

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RTP specified in the CORE OPERATING LIMITS REPORT (COLR), and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature N-16 trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature N-16 trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735) psig of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

PROOF & REVIEW

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

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

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

1

ACTION:

1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2)

With the SHUTDOWN MARGIN less than ~~[1.6]% Δk/k~~, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~[1.6]% Δk/k~~:

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

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

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

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

*See Special Test Exceptions Specification 3.10.1.

PROOF & REVIEW

REACTIVITY CONTROL SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to [1.3%] Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than [1.3%] Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

3

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to [1.3%] Δk/k:

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

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

PROOF & REVIEW

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid storage tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the RCS.

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

ACTION:

(3) With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path from the boric acid storage tanks is greater than or equal to 65°F when it is a required water source;

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; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

---

*A maximum of two charging pumps shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F except when Specification 3.4.8.3 is not applicable. An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve(s) with power removed from the valve operator(s) or by a manual isolation valve(s) secured in the closed position.

Proof & Review

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 indicated borated water level of 50%,
2) A minimum boron concentration of 7000 ppm, and
3) A minimum solution temperature of 65°F.

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

1) A minimum indicated borated water level of 95%,
2) A boron concentration between 2000 ppm and 2200 ppm,
3) A minimum solution temperature of 40°F, and
4) A maximum solution temperature of 120°F.

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

ACTION:

3

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 equivalent to at least 1.3% Δk/k 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.

TABLE 3

REACTOR TRIP SYSTEM

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3[a], 4[a], 5[a]	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1[c], 2	2
3. Power Range, Neutron Flux, High Positive Rate	4	2	3	1, 2	2
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5. Intermediate Range, Neutron Flux	2	1	2	1[c], 2	3
6. Source Range, Neutron Flux					
a. Reactor Trip and Indication					
1) Startup	2	1	2	2[b]	4
2) Shutdown	2	1	2	3, 4, 5	5.1
b. Boron Dilution Flux Doubling*	2	1	2	3[h], 4, 5	5.1, 5.2
7. Overtemperature N-16	4	2	3	1, 2	12
8. Overpower N-16	4	2	3	1, 2	12
9. Pressurizer Pressure--Low	4	2	3	1[d]	6[e]
10. Pressurizer Pressure--High	4	2	3	1, 2	6

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3 and for Unit 2 six months after initial criticality.

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TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3 and for Unit 2 six months after initial criticality.

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7						
1) Power Range Neutron Flux P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
2) Turbine First Stage Pressure P-13	N.A.	R	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
2 e. ~~d.~~ Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3^a, 4^a, 5^a
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3^a, 4^a, 5^a
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(15), R(16)	N.A.	1, 2, 3^a, 4^a, 5^a

PROOF & REVIEW

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
7. Automatic Initiation of ECCS Switchover to Containment Sump (Continued)					
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	26
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Loss of Power (6.9 kV & 480 V Safeguards System Undervoltage)					
a. 6.9 kV Preferred Offsite Source Undervoltage	2/bus	2/bus	1/bus	$1^f, 2^f, 3^f, 4^f$	23
b. 6.9 kV Alternate Offsite Source Undervoltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
c. 6.9 kV Bus Undervoltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
d. 6.9 kV Degraded Voltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
e. 480 V Degraded Voltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
f. 480 V Low Grid Undervoltage	2/bus	2/bus	1/bus	1, 2, 3,	23
9. Control Room Emergency Recirculation					
a. Manual Initiation	2	1	2	All	24

PROOF & REVIEW

TABLE 3.3-2 (Continued)

②

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	18
b. Reactor Trip, P-4	2	2	2	1, 2, 3	20
11. Solid State Safeguards Sequencer (SSSS)					
a. Safety Injection Sequence	1/train	1/train	1/train	1, 2, 3, 4	12
b. Blackout Sequence	1/train	1/train	1/train	1, 2, 3, 4	25

9. Control Room Emergency Recirculation (Continued)

PROOF & REVIEW

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (ECCS, Reactor Trip, Feedwater Isolation, Control Room Emergency Recirculation, Emergency Diesel Generator Operation, Containment Vent Isolation, Station Service Water, Phase A Isolation, Auxiliary Feedwater-Motor Driven Pump, Turbine Trip, Component Cooling Water, Essential Ventilation Systems, and Containment Spray Pump).					
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 1	2.7	0.71	1.7	≤ 3.2 psig	≤ 3.8 psig
2 d. Pressurizer Pressure--Low					
1) Unit 1	15.0	10.91	2.0	≥ 1820 psig	≥ 1803.6 psig
2) Unit 2	15.0	11.3	2.0	≥ 1820 psig	≥ 1803.6 psig
e. Steam Line Pressure--Low					
1) Unit 1	17.3	15.01	2.0	≥ 605 psig*	≥ 593.5 psig*
2) Unit 2	17.3	9.15	2.0	≥ 605 psig*	≥ 578.4 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	2.7	0.71	1.7	≤ 18.2 psig	≤ 18.8 psig

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
c. Containment Vent Isolation								
1) Manual Initiation	See Item 3.a.1 and 2.a above. Containment vent isolation is manually initiated when Phase "A" isolation function or containment spray function is manually initiated.							1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
4 d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	[illegible], 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	[illegible]
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	[illegible], 2

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Automatic Initiation of ECCS Switchover to Containment Sump (Continued)								
b. RWST Level-Low-Low Coincident With	S	SR	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Loss of Power (6.9 kV & 480 V Safeguards System Undervoltage)								
a. 6.9 kV Preferred Offsite Source Undervoltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 6.9 kV Alternate Offsite Source Undervoltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
c. 6.9 kV Bus Undervoltage	N.A	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
d. 6.9 kV Degraded Voltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
e. 480 V Degraded Voltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
f. 480 V Low Grid Undervoltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4

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PROOF & REVIEW

INSTRUMENTATION

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The explosive gas monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded.

APPLICABILITY: ~~As shown in Table~~ 3.3-7.

ACTION: (Units 1 and 2)

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-7.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-7. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each explosive gas monitoring instrumentation channel shown in Table 3.3-7 shall be demonstrated OPERABLE:

a. At least once per 24 hours by performance of a CHANNEL CHECK,

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

c. At least once per 92 days by performance of a CHANNEL CALIBRATION which shall include the use of standard gas samples in accordance with the manufacturer's recommendations.

PROOF & REVIEW

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam line inoperable and/or with one stop valve or one control valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required overspeed protection system shall be demonstrated OPERABLE:

7 a. At least once per ~~14 days~~ 6 weeks by cycling each of the following valves through at least one complete cycle from the running position using the manual test or Automatic Turbine Tester (ATT):

1) Four high pressure turbine stop valves,
2) Four high pressure turbine control valves,
3) Four low pressure turbine stop valves, and
4) Four low pressure turbine control valves.

b. At least once per 14 days by testing of the two mechanical overspeed devices using the Automatic Turbine Tester or manual test

c. At least once per ~~31 days~~ 6 weeks by direct observation of the movement of each of the above valves through one complete cycle from the running position.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats (if applicable), disks and stems and verifying no unacceptable flaws. If unacceptable flaws are found, all other valves of that type shall be inspected.

*Not applicable in MODES 2 and 3 with all main steam line isolation valves and associated bypass valves in the closed position.

PROOF & REVIEW

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

2 LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With no loop 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 to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), and/or RHR pump(s) if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE

a. By verifying secondary side water level to be greater than or equal to 10% (narrow range) at least once per 12 hours, and

b. By performing the surveillances pursuant to Specification 4.0.6.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

PROOF & REVIEW

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the Reactor Coolant System Leakage Detection System required by Specification 3.4.5.1 at least once per 12 hours;

b. 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 with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

c. Performance of a Reactor Coolant System water inventory balance at least within 12 hours after achieving steady state operation* and at least once per 72 hours thereafter during steady state operation, except that no more than 96 hours shall elapse between any two successive inventory balances. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4; and

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

4.4.5.2.2 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 per 18 months,

b. 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 8701A, 8701B, 8702A, and 8702B.**

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following check valve actuation due to flow through the valve.

5 ~~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.

*T_{avg} being changed by less than 5°F/hour.

**This exception allowed since these valves have control room position indication, inadvertent opening interlocks and a system high pressure alarm.

PROOF & REVIEW

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2) A visual inspection of the containment 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.

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

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pumps,

b) Safety injection pumps, and

c) RHR pumps.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1)	Centrifugal charging pump	$\geq$ 2370 psid,
2)	Safety injection pump	$\geq$ 1440 psid, and
3)	RHR pump	> 170 psid.

g. By verifying the correct position of each mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2

2) At least once per 18 months:

CCP/SI System Valve Number	SI System Valve Number	
SI-8810A	SI-8822A	SI-8816A
SI-8810B	SI-8822B	SI-8816B
SI-8810C	SI-8822C	SI-8816C
SI-8810D	SI-8822D	SI-8816D

PROOF & REVIEW

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

(2) CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 2.1.1 of the Technical Requirements Manual.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 48.3 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days. The blind flange on the fuel transfer canal need not be verified closed except after each drainage of the canal.

PROOF & REVIEW

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.#

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

ACTION:

*With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

4 #The requirements of Specification 3.6.3 do not apply for those valves covered by Specifications 3.7.1.1, 3.7.1.5, and 3.7.1.6, and 3.7.1.7.

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

PROOF & REVIEW

PLANT SYSTEMS

3/4.7.4 STATION SERVICE WATER SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent station service water loops per unit and the cross-connect between the Station Service Water Systems of each unit shall be OPERABLE.

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

5 ACTION:

a. With only one station service water loop per unit OPERABLE, restore at least two loops per unit to OPERABLE status within 72 hours or for the unit(s) with the inoperable station service water loop be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more of the cross-connects ~~or cross-connect valve(s)~~ inoperable, within 7 days ~~72 hours~~ restore the cross-connects to OPERABLE status~~, or open the affected valve(s), and maintain the affected valve(s) open;~~ otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1.1 Each station service water loop shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, ~~sealed~~, or otherwise secured in position is in its correct position; and

2

b. At least once per 18 months during shutdown, by verifying that each station service water pump starts automatically on a Safety Injection test signal.

4.7.4.1.2 At least once per 92 days the cross-connects shall be demonstrated OPERABLE by cycling the cross-connect valves or verifying that the valves are locked open.

PROOF & REVIEW

PLANT SYSTEMS

STATION SERVICE WATER SYSTEM

ONE UNIT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.4.2 At least two independent station service water loops in the operating unit*, at least one station service water pump in the shutdown unit** and the cross-connects from the OPERABLE station service water pump(s) in the shutdown unit to the station service water loops of the operating unit shall be OPERABLE.

APPLICABILITY: Unit 1 (Unit 2) in MODES 1, 2, 3 and 4
Unit 2 (Unit 1) in MODES 5, 6 and Defueled

ACTION:

a. With one station service water loop in the operating unit inoperable, restore two loops in the operating unit 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.

5 b. With one or more of the cross-connects ~~or cross-connect valve(s)~~ between the OPERABLE station service water pump(s) in the shutdown unit and the station service water loops in the operating unit inoperable, within 7 days ~~72 hours~~ restore the cross-connects to OPERABLE status ~~or open the affected valve(s) and maintain the affected valve(s) open.~~ Otherwise, place the operating unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. If neither station service water pump in the shutdown unit is OPERABLE, restore at least one pump to OPERABLE status within ~~72 hours~~ 7 days or place the operating unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.2.1 Each station service water loop in the operating unit shall be demonstrated OPERABLE per the requirements of Specification 4.7.4.1.1.

4.7.4.2.2 At least once per 92 days the cross-connect(s) between the OPERABLE station service water pump(s) in the shutdown unit and the station service water loops in the operating unit shall be demonstrated OPERABLE by cycling the cross-connect valves in the flow path or verifying that these valves are locked open.

* A Unit in MODE 1, 2, 3 or 4 is designated as the "Operating unit".
** A unit is MODE 5, 6 or Defueled is designated as the "shutdown unit".

PROOF & REVIEW

PLANT SYSTEMS

(2) SURVEILLANCE REQUIREMENTS (Continued)

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 filtration unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% by using the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the emergency filtration unit flow rate is 8000 cfm ± 10%, and the emergency pressurization unit flow rate is 800 cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*, for a methyl iodide penetration of less than 0.2%; and

   3) Verifying an emergency filtration unit flow rate of 8000 cfm ± 10% and an emergency pressurization unit flow rate of 800 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980;

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*, for a methyl iodide penetration of less than 0.2%;

*ANSI N510-1980 and ANSI N509-1980 shall be used in place of ANSI N510-1975 and ANSI N509-1976, respectively.

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PLANT SYSTEMS

CONTROL ROOM HVAC SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.7.2 Two independent control room HVAC trains shall be OPERABLE.

APPLICABILITY: MODES 5 and 6:

ACTION:

a. With one control room HVAC train inoperable, restore the inoperable train to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room HVAC train in the emergency recirculation mode.

2 b. With both control room HVAC trains inoperable, or with the OPERABLE control room HVAC train required to be in the emergency recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.7.2 Each control room HVAC train shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.7.7.1.

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TABLE 3.7-3

AREA TEMPERATURE MONITORING

AREA	MAXIMUM TEMPERATURE LIMIT (°F) Normal Conditions	Abnormal Conditions
1. Electrical and Control Building		
Normal Areas	104	131
Control Room Main Level (El. 830'-0")	80	104
Control Room Technical Support Area (El. 840'-6")	104	104
UPS/Battery Rooms	104	113
Chiller Equipment Areas	122	131
2. Fuel Building		
Normal Areas	104	131
Spent Fuel Pool Cooling Pump Rooms	122	131
3. Safeguards Buildings		
Normal Areas	104	131
AFW, RHR, SI, Containment Spray Pump Rooms	122	131
RHR Valve and Valve Isolation Tank Rooms	122	131
RHR/CT Heat Exchanger Rooms	122	131
Diesel Generator Area	122	131
Diesel Generator Equipment Rooms	130	131
Day Tank Room	122	131
4. Auxiliary Building		
4 Normal Areas	104	131
~~AFW RHR, SI Containment Spray~~ CCW, CCP Pump Rooms	122	131
CCW Heat Exchanger Area	122	131
CVCS Valve and Valve Operating Rooms	122	131
Auxiliary Steam Drain Tank Equipment Room	122	131
Waste Gas Tank Valve Operating Room	122	131
5. Service Water Intake Structure	127	131
6. Containment Buildings		
General Areas	120	129
Reactor Cavity Exhaust	150	190
CRDM Shroud Exhaust	163	172

PROOF & REVIEW

(2) LIMITING CONDITION FOR OPERATION (Continued)

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 4.8.1.1.1a. 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 Requirement 4.8.1.1.2a.4) 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 the 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 Requirement 4.8.1.1.2a.4) 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.

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 Surveillance Requirement 4.8.1.1.2a.4) 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. With only one

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

#During performance of surveillance activities as a requirement for ACTION statements, the air-roll test shall not be performed.

PROOF & REVIEW

ELECTRICAL POWER SYSTEMS

(2) LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) the 6.9 kV safeguards bus power supply from the preferred offsite source to the alternate offsite source.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day fuel tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 441 rpm in less than or equal to 10 seconds.*

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*All planned diesel engine starts for the purpose of this surveillance may be preceded by a prelube period in accordance with vendor recommendations.

Proof
&
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RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

2 3.11.1 The quantity of radioactive material contained in each unprotected outdoor tank shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any unprotected outdoor tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1 The quantity of radioactive material contained in each of the unprotected outdoor tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this specification are those unprotected outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

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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.

2

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APPLICABILITY

BASES

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours.

PROOF & REVIEW

APPLICABILITY

BASES

co ivention of identifying valves; without the unit designator if the remainder o the tag number is applicable to both units, with the unit designator if the tag is only applicable to one unit.

2 When a specification is shared per 3.0.5c the ACTION section contains the identifier "(Units 1 and 2)".

Specifications 4.0.1 through 4.0.6 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establis .s the requirement that surveillances must be met during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verificaion of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

PROOF & REVIEW

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

1.6% Δk/k for Unit 1 (1.3% Δk/k for Unit 2)

(2) SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~1.6% Δk/k~~ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than (3) 200°F, a SHUTDOWN MARGIN of 1.3% Δk/k provides adequate protection and is based on the results of the boron dilution accident analysis.

Since the actual overall core reactivity balance comparison required by 4.1.1.1.2 cannot be performed until after criticality is attained, this comparison is not required (and the provisions of Specification 4.0.4 are not applicable) for entry into any Operational Mode within the first 31 EFPD following initial fuel load or refueling.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

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

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

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

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REACTIVITY CONTROL SYSTEMS

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

2 This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

1.6% $\Delta k/k$ for Unit 1 (1.3% $\Delta k/k$ for Unit 2)

3 With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of [1.6% $\Delta k/k$] after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires [15,700] gallons of 7000 ppm borated water from the boric acid storage tanks or [70,702] gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

PROOF & REVIEW

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

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

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

The limitation for minimum solution temperature of the borated water sour-- are sufficient to prevent boric acid crystallization with the highest allowab; boron concentration.

3 The boron capability required below 200°F is sufficient to provide a SHU-DOWN MARGIN of [1.30]% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either [1,100] gallons of 7000 ppm borated water from the boric acid storage tanks or [7,113] gallons of 2000 ppm borated water from the RWST.

As listed below, the required indicated levels for the boric acid storage tanks and the RWST include allowances for required/analytical volume, unusable volume, measurement uncertainties (which include instrument error and tank tolerances, as applicable), system configuration requirements, and other required volume.

Tank	MODES	Ind. Level	Unusable Volume (gal)	Required Volume (gal)	Measurement Uncertainty	System Config. (gal)	Other (gal)
RWST	5,6	[24%	45,494	7,113	4% of span	57,857	N/A
	1,2,3,4	95%	45,494	70,702	4% of span	N/A	357,535*
Boric Acid Storage Tank	5,6	10%	3,221	1,100	6% of span	N/A	N/A
	5,6 (gravity feed)	20%	3,221	1,100	6% of span	3,679	N/A
	1,2,3,4	50%	3,221	15,700	6% of span	N/A	N/A]

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

*Additional volume required to meet Specification 3.5.4.

PROOF & REVIEW

POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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;

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

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

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

2 Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset ~~this~~ reduction in the generic margin. The DNBR generic margin, totaling 9.1% for Unit 1 and 10.1% for typical cells and 9.5% for thimble cells for Unit 2, ~~for DNBR~~ completely offset any rod bow penalties. This margin includes the following for Unit 1:

a. Design limit DNBR of 1.30 vs 1.28,

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

c. Thermal Diffusion Coefficient of 0.038 vs 0.051,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The margin for Unit 2 is included by establishing a fixed difference between the safety analysis limit DNBR and the design limit DNBR.

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

PROOF & REVIEW

POWER DISTRIBUTION LIMITS

BASES

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR at or above the safety analysis limit value throughout each analyzed transient. The Unit 1 indicated T_{avg} value of 592.7°F (conservatively rounded to 592°F) and the Unit 1 indicated pressurizer pressure value of 2207 psig correspond to analytical limits of 594.7°F and 2193 psig respectively, with allowance for measurement uncertainty. The Unit 2 indicated T_{avg} value of 592.8°F (conservatively rounded to 592°F) and the Unit 2 indicated pressurizer pressure value of 2219 psig correspond to analytical limits of 595.16°F and 2205 psig respectively, with allowance for measurement uncertainty. The indicated uncertainties assume that the reading from four channels will be averaged before comparing with the required limit.

2

align

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation, and to detect any significant flow degradation of the Reactor Coolant System (RCS).

The additional surveillance requirements associated with the RCS total flow rate are sufficient to ensure that the measurement uncertainties are limited to 1.8% as assumed in the Improved Thermal Design Procedure Report for CPSES.

Performance of a precision secondary calorimetric is required to precisely determine the RCS temperature. The transit time flow meter, which uses the N-16 system signals, is then used to accurately measure the RCS flow. Subsequently, the RCS flow detectors (elbow tap differential pressure detectors) are normalized to this flow determination and used throughout the cycle.

PROOF & REVIEW

PLANT SYSTEMS

BASES

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

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

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 430 gpm to two steam generators at a pressure of 1221 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 860 gpm to four steam generators at a pressure of 1221 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

The Auxiliary Feedwater System is capable of delivering a total feedwater flow of 430 gpm at a pressure of 1221 psig to the entrance of at least two steam generators while allowing for: (1) any possible spillage through the design worst case break of the main feedwater line; (2) the design worst case single failure; and (3) recirculation flow. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed in operation. The test flow for the steam-driven auxiliary feedwater pump at a pressure of greater than or equal to 1450 psid ensures this capability.

The auxiliary feedwater flow path is a passive flow path based on the fact that valve actuation is not required in order to supply flow to the steam generators. The automatic valves tested in the flow path are the Feedwater Split Flow Bypass which are required to be shut upon initiation of the Auxiliary Feedwater System to meet the requirements of the accident analysis.

Both steam supplies for the turbine-driven auxiliary feedwater pump must be OPERABLE in order to meet the design bases for the complete range of accident analyses. The allowed outage time for one inoperable steam source is consistent with the lower probability of the worst case steam or feedwater line break accident.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 18 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power or 4 hours at HOT STANDBY followed by a cooldown to 350°F at a rate of 50°F/hr for 5 hours. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The required indicated level includes a 3.5-percent measurement uncertainty, an unusable volume of 12,100 gallons and a required usable volume of ~~249,000~~ gallons. — 249,900

1

NUREG-0737, Item II.E.1.1 requires a backup source to the CST which is the CPSES Station Service Water System, which can be manually aligned, if required in lieu of CST minimum water volume.

PROOF & REVIEW

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential of 5 psid with respect to the outside atmosphere, and (2) the containment peak pressure does not exceed the design pressure of 50 psig during a LOCA.

(3) The indicated containment pressure values of [-0.3] psig and [1.3] psig correspond to analytical limits of [-0.5] psig and 1.5 psig, respectively, with allowance for measurement uncertainty.

The maximum peak pressure expected to be obtained from a LOCA event is [48.3] psig, which is less than design pressure and is consistent with the safety analyses. This value includes the limit of 1.5 psig for initial positive containment pressure.

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 LOCA or steam line break accident. The average temperature shall be by an adjusted averaging of at least 2 of the measurements made at the listed locations, by fixed or portable instruments with allowance for temperature measurement uncertainty.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of 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 48.3 psig in the event of a LOCA. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch and 12-inch containment and hydrogen purge supply and exhaust isolation valves are required to be locked 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 locked closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Ventilation System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Containment Ventilation System during operations is restricted to the 18-inch pressure relief discharge isolation valves (with an effective diameter of 3 inches) since, these venting valves are capable of closing during a LOCA or steam line break accident. Therefore, the Exclusion Area dose guideline of 10 CFR 100 would not be exceeded in the event of an accident during containment venting operation.

TABLE B 3/4.4-1b

UNIT 2 REACTOR VESSEL FRACTURE TOUGHNESS PROPERTIES

COMPONENT	GRADE	Code NO.	Cu %	Ni %	P %	T_{NDT} °F	T_{CV} 50 FT-LB 35 MIL TEMP. °F	RT_{NDT} °F	AVG. SHELF ENERGY MWD(b) FT-LB	AVG. SHELF ENERGY NMWD(c) FT-LB
Closure Hd. Dome	A533B, Cl.1	R3811-1	0.15	.65	0.014	-40	60	0	-	131
Closure Hd. Torus	A533B, Cl.1	R3810-1	0.15	.69	0.011	-30	30	-30	-	143
Closure Hd. Flange	A508, Cl.2	R3802-1	-	.71	0.013	40	<100	40	-	152
Vessel Flange	A508, Cl.2	R3801-1	-	.70	0.009	-10	<50	-10	-	121
Inlet Nozzle	A508, Cl.2	R3803-1	-	.84	0.009	-10	<50	-10	-	138
Inlet Nozzle	A508, Cl.2	R3803-2	0.10	.91	0.008	-20	<40	-20	-	136
Inlet Nozzle	A508, Cl.2	R3803-3	-	.91	0.010	-10	<50	-10	-	146
Inlet Nozzle	A508, Cl.2	R3803-4	-	.86	0.009	-20	<40	-20	-	136
Outlet Nozzle	A508, Cl.2	R3805-1	-	.64	0.006	0	<60	0	-	132
Outlet Nozzle	A508, Cl.2	R3805-2	-	.66	0.005	0	<60	0	-	119
Outlet Nozzle	A508, Cl.2	R3805-3	-	.66	0.004	0	~~<60~~	0	-	117
Outlet Nozzle	A508, Cl.2	R3805-4	-	.67	0.005	0	~~>60~~ <60	0	-	119
Nozzle Shell	A533B, Cl.1	R3806-1	0.05	.61	0.010	-10	100	40	-	76
Nozzle Shell	A533B, Cl.1	R3806-2	0.06	.62	0.009	-30	70	10	-	87
Nozzle Shell	A533B, Cl.1	R3806-3	0.06	.70	0.007	-30	100	40	-	86
Inter. Shell	A533B, Cl.1	R3807-1	0.06	.64	0.006	-20	<40	-20	133	108
Inter. Shell	A533B, Cl.1	R3807-2	0.06	.64	0.007	-20	70	10	1[illegible]2	101
Inter. Shell	A533B, Cl.1	R3807-3	0.05	.60	0.007	-20	40	-20	120	105
Lower Shell	A533B, Cl.1	R3816-1	0.05	.59	0.001	-30	30	-30	136	107
Lower Shell	A533B, Cl.1	R3816-2	0.03	.65	0.002	-30	60	0	131	106
Lower Shell	A533B, Cl.1	R3816-3	0.04	.63	0.008	-40	20	-40	139	[illegible]
Bottom Hd. Torus	A533B, Cl.1	R3813-1	0.12	.65	0.009	-60	0	-60	-	123
Bottom Hd. Dome	A533B, Cl.1	R3814-1	0.12	.66	0.009	-70	-10	-70	-	[illegible]
Weld Metal (a) (Inter. to Lower Shell Girth Seam)			0.05	.03	0.004	-60	[illegible]0	-60	-	86
Weld Metal (b) (Inter. & Lower Shell Long Seams)			0.07	.05	0.005	-50	[illegible]10	-50	-	[illegible]

(a) B4 Weld Wire Ht. 89833 & Linde 124 Flux Lot No. 1061
(b) B4 Weld Wire Ht. 89833 & Linde 0091 Flux Lot No. 1054
(c) Normal to major working direction
(d) Major working direction

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ELECTRICAL POWER SYSTEMS

BASES

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

The Fuel Storage System consists of the fuel oil storage tank and is equivalent to the ANSI N195-1976 definition for supply tank.

2 The Surveillance Requirement for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971; 1.108, "Periodic Testing 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," January 1978, Generic Letter 84-15, and Generic Letter 83-26, "Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests."

The Diesel Generator Test schedule, Table 4.8-1, is based on the recommendations of Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and NRC Technical Report A-3230, "Evaluation of Diesel Unavailability and Risk Effective Surveillance Test Intervals," May 1986, and Generic Letter 84-15, "Proposed Staff Position to Improve and Maintain Diesel Generator Reliability."

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," Revision 1, February 1978, Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," Revision 2, February 1977, and IEEE STD 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

The operational requirement to energize the instrument busses from their associated inverters connected to its associated D.C. bus is satisfied only when the inverter's output is from the regulated portion of the inverter and not from the unregulated bypass source via the internal static switch.

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.

PROOF & REVIEW

ELECTRICAL POWER SYSTEMS

BASES

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

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 2.07 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. This is based on the recommendations of Regulatory Guide 1.63, Revision 2, July 1978, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants."

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The Surveillance Requirements applicable to lower voltage circuit breakers ~~and fuses~~ provide assurance of breaker reliability by testing at least 10% 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.

All Class 1E motor-operated valves' motor starters are provided with thermal overload protection which is permanently bypassed and provides an alarm function only at Comanche Peak Steam Electric Station. Therefore, there are no OPERABILITY or Surveillance Requirements for these devices, since they will not prevent safety-related valves from performing their function (refer to Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977).

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SECTION 5.0

DESIGN FEATURES

(2)

COMANCHE PEAK - UNITS 1 AND 2

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REVIEW

SECTION 6.0

ADMINISTRATIVE CONTROLS

2

PROOF & REVIEW

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
SS	1	1	1
SRO	1	None**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	None	1***

SS - Shift Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary 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 Shift Supervisor from the control room while the 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 Shift Supervisor from the control room while the 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.

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

2 **At least one licensed Senior Operator or licensed Senior Operator limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications described in Option 1 of the Commission Policy Statement on Engineering Expertise (50 FR 43621, October 28, 1985).

PROOF & REVIEW

TECHNICAL REVIEW AND CONTROLS (Continued)

department manager as previously designated by the Vice President, Nuclear Operations, in writing. Individuals responsible for procedure reviews shall be members of the Nuclear Operations Management Staff previously designated by the Vice President, Nuclear Operations. Changes to procedures which do not change the intent of approved procedures may be approved for implementation by two members of the Nuclear Operations Management Staff, at least one of whom holds a Senior Operator License, provided such approval is prior to implementation and is documented. Such changes shall be approved by the original approval authority within 14 days of implementation;

b. Proposed tests and experiments which affect plant nuclear safety shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by a qualified individual/group other than the individual/group which prepared the proposed test or experiment. Proposed test and experiments shall be approved before implementation by the Plant Manager. Individuals responsible for conducting such reviews shall be members of the Nuclear Operations Management Staff previously designated by the Vice President, Nuclear Operations;

c. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Chief Engineer. Each such modification shall be reviewed by a qualified individual/group meeting the experience requirements of ANSI N18.1-1971, Section 4.6 other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Individuals/groups responsible for conducting such reviews shall be previously designated by the Chief Engineer. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Plant Manager prior to implementation;

d. Individuals responsible for reviews performed in accordance with the requirements of Specifications 6.5.3.1a and 6.5.3.1b, shall be members of the Nuclear Operations Management staff previously designated by the Vice President, Nuclear Operations. Each such review shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be done in accordance with the appropriate qualification requirements;

e. Each review shall include a determination of whether or not an unreviewed safety question is involved. For items involving unreviewed safety questions, NRC approval shall be obtained prior to the Plant Manager's approval for implementation; and

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f. The Security Plan and Emergency Plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes to the implementing procedures shall be approved by the Vice President,

PROOF & REVIEW

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program (Continued)

2 be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR 50, and

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MONTHLY OPERATING REPORTS (Continued)

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shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, 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.

CORE OPERATING LIMITS REPORT

6.9.1.6a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1). Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2). Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

3). Control Rod Insertion Limits for Specification 3/4.1.3.6,

4). AXIAL FLUX DIFFERENCE Limits and target band for Specification 3/4.2.1.,

5). Heat Flux Hot Channel Factor, K(Z), W(Z), and F_Q^{RTP} for Specification 3/4.2.2,

6). Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

6.9.1.6b The analytical methods used to determine the core operating limits are for Units 1 and 2, unless otherwise stated, and shall be those previously approved by the NRC in:

1). WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2. - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2). WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary). (Methodology for Specification 3.2.1. - Axial Flux Difference, [Constant Axial Offset Control].)

3). T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

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4). NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

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RECORD RETENTION (Continued)

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the Quality Assurance Manual;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the SORC and the ORC;

l. Records of the service lives of all hydraulic and mechanical snubbers required by the Technical Requirements Manual including the date at which the service life commences and associated installation and maintenance records;

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m. Records of secondary water sampling and water quality; and

n. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 10 CFR 20.203(c)(5), in lieu of the "control device" or "alarm signal" required by paragraph 10 CFR 20.203(c), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be

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6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3o. This documentation shall contain:

2 1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the SORC and the approval of the Vice President, Nuclear Operations.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM.

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3o. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the SORC and the approval of the Vice President, Nuclear Operations.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.