Docket Nos. 50-275
and 50-323

August 6, 1992

Mr. Gregory M. Rueger
Senior Vice President and General Manager
Nuclear Power Generation Business Unit
Pacific Gas and Electric Company
77 Beale Street, Room 1451
San Francisco, California 94106

Dear Mr. Rueger:

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M82966) AND UNIT NO. 2 (TAC NO. M82967)

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-80 and Amendment No. 71 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated February 4, 1992 (reference License Amendment Request LAR 92-02).

These amendments revise the combined technical specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to remove cycle specific information that is no longer applicable. Also, TS Table 3.3-5 has been revised to correct the table notation which was incorrectly modified by License Amendments 51 and 50.

A copy of the related Safety Evaluation is enclosed. A notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original signed by
Harry Rood, Senior Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 72 to DPR-80
2. Amendment No. 71 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

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DATE	7/7/92	7/8/92	7/9/92	7/16/92	8/6/92

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

August 6, 1992

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1. Amendment No.72 to DPR-80
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cc w/enclosures:
See next page

Mr. Gregory M. Rueger
Pacific Gas and Electric Company

Diablo Canyon

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
License No. DPR-80

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

   A. The application for amendment by Pacific Gas & Electric Company (the licensee) dated February 4, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 72 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harry Rood for

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 6, 1992

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. DPR-82

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

   A. The application for amendment by Pacific Gas & Electric Company (the licensee) dated February 4, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 71 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harry Rood for

Theodore R. Quay, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 6, 1992

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages are also included, as appropriate.

REMOVE	INSERT
ix	ix
xii	xii
2-2a	---
2-4	2-4
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
B 2-7	B 2-7
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-7	3/4 1-7
3/4 1-8	3/4 1-8
3/4 1-9	3/4 1-9
3/4 1-12	3/4 1-12
3/4 1-13	3/4 1-13
3/4 1-20	3/4 1-20
3/4 1-23	---
3/4 1-24	---
3/4 2-22	3/4 2-22
3/4 3-3	3/4 3-3
3/4 3-8	3/4 3-8
3/4 3-11	3/4 3-11
3/4 3-23	3/4 3-23
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-31a	---
3/4 4-8	3/4 4-8
3/4 5-1	3/4 5-1
3/4 5-6	3/4 5-6
3/4 5-9	3/4 5-9
3/4 5-10	3/4 5-10
3/4 7-9a	3/4 7-9a
3/4 9-1	3/4 9-1

REMOVE	INSERT
3/4 10-1	3/4 10-1
B 3/4 1-2	B 3/4 1-2
B 3/4 5-2a	B 3/4 5-2a

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. Low Setpoint	≤ 25% of RATED THERMAL POWER	≤ 26% of RATED THERMAL POWER
b. High Setpoint	≤ 109% of RATED THERMAL POWER	≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure-Low	≥ 1950 psig	≥ 1940 psig
10. Pressurizer Pressure-High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level-High	≤ 92% of instrument span	≤ 93% of instrument span
12. Reactor Coolant Flow-Low	≥ 90% of minimum measured flow** per loop	≥ 88.9% of minimum measured flow** per loop

**Minimum measured flow is 89,800 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	≥ 7.2% of narrow range instrument span-each steam generator	≥ 6.2% of narrow range instrument span-each steam generator
14. DELETED		
15. Undervoltage-Reactor Coolant Pumps	≥ 8050 volts-each bus	≥ 7935 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 54.0 Hz - each bus	≥ 53.9 Hz - each bus
17. Turbine Trip		
a. Low Autostop Oil Pressure	≥ 50 psig	≥ 45 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	N.A.	N.A.
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
20. Reactor Trip Breakers	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
22. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
2) P-13 Input	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	≤ 35% of RATED THERMAL POWER	≤ 36% of RATED THERMAL POWER
d. Power Range Neutron Flux, P-9	≤ 50% of RATED THERMAL POWER	≤ 52.1% of RATED THERMAL POWER
e. Power Range Neutron Flux, P-10	10% of RATED THERMAL POWER	≥ 9%, ≤ 11% of RATED THERMAL POWER
f. Turbine Impulse Chamber Pressure, P-13	≤ 10% RTP Turbine Impulse Pressure Equivalent	≤ 11% RTP Turbine Impulse Pressure Equivalent
23. Seismic Trip	≤ 0.35 g	≤ 0.40 g

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

P = Pressurizer pressure, psig;

P' = 2235 psig (indicated RCS nominal operating pressure);

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 30 s, τ_2 = 4 s;

S = Laplace transform operator, s^{-1};

K_1 = 1.200

K_2 = 0.01817/°F

K_3 = 0.000831/psig

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 1 (Continued)

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t - q_b between - 19% and + 9%, f_1 (ΔI) = 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 - 19%, the ΔT Trip Setpoint shall be automatically reduced by 2.75% of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of (q_t - q_b) exceeds + 9%, the ΔT Trip Setpoint shall be automatically reduced by 1.76% 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 3.2%.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where: ΔT_0 = Indicated ΔT at rated power;

T = Average temperature, $^{\circ}F$;

T'' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

K_4 = 1.072

K_5 = 0.0174/°F for increasing average temperature and 0 for decreasing average temperature;

K_6 = 0.00145/°F for $T > T''$; $K_6 = 0$ for $T \leq T''$;

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 s;

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.6%.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 seconds. On decreasing power, the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power, the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position trip is an anticipatory trip which provides score protection against DNB. The Open/Close Position trip assures a reactor trip signal is generated before the Low Flow Trip Setpoint is reached. No credit was taken in the safety analyses for operation of this trip. The functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Trip System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent) an automatic reactor trip will occcur if more than one reactor coolant pump breaker is opened. Below P-7 the trip function is automatically blocked.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

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

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

3/4.1 REACTIVITY CONTROL SYTEMS

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.

ACTION:

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 MODES 1 or 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.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, 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.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% 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

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

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

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than $+5 \times 10^{-5}$ $\Delta k/k/^{\circ}F$ for 0% to 70% RATED THERMAL POWER, and for > 70% to 100% RATED THERMAL POWER the MTC decreases linearly to 0 $\Delta K/\bar{K}/^{\circ}F$ for the all rods withdrawn condition, beginning of cycle life (BOL); or

b. Less negative than -3.9×10^{-4} $\Delta k/k/^{\circ}F$ for all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

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

ACTION:

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

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limit of Specification 3.1.1.3a within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

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

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

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

c. The provisions of Specification 3.0.4 are not applicable.

---

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

#See Special Test Exceptions Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE with motor-operated valves required to change position and pumps required to operate for boron injection capable of being powered from an OPERABLE emergency power source:

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the flow path is greater than or equal to 65°F when a flow path from the boric acid tanks is used, and

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. The flow path from the refueling water storage tank via a charging pump to the RCS.

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

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path 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.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each 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 tanks is greater than or equal to 65°F,

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,

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

d. 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.

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 323°F.

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CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pump is a centrifugal charging pump, verify that, on recirculation flow, the centrifugal charging pump develops a differential pressure of greater than or equal to 2400 psid.

4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours, except when the reactor vessel head is removed, by verifying that the motor breaker D.C. control power is de-energized.

*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pumps include a centrifugal charging pump(s), verify that, on recirculation flow, each required centrifugal charging pump(s) develops a differential pressure of greater than or equal to 2400 psid.

4.1.2.4.2 All centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to 323°F by verifying that the motor breaker D.C. control power is de-energized.

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 323°F.

*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 2,499 gallons,
2) A boron concentration between 7,000 and 7,700 ppm, and
3) A minimum solution temperature of 65°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A minimum contained borated water volume of 50,000 gallons,
2) A minimum boron concentration of 2300 ppm, and
3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water source(s) shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 14,042 gallons,
2) A boron concentration between 7,000 and 7,700 ppm, and
3) A minimum solution temperature of 65°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A contained borated water volume of greater than or equal to 400,000 gallons,
2) A boron concentration between 2300 and 2500 ppm, and
3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the boron concentration in the water,

   2) Verifying the contained borated water volume of the water source, and

   3) Verifying the Boric Acid Storage System solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 35°F.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

#See Special Test Exceptions Specification 3.10.4

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
Actual Reactor Coolant System T_{avg}	$\leq$ 584.3°F
Actual Pressurizer Pressure	$\geq$ 2212 psia*

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop	2/loop in one loop	2/loop in each loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two loops	2/loop in each loop	1	6
13. Steam Generator Water Level-Low-Low	3/S.G.	2/S.G. in one S.G.	2/S.G. in each S.G.	1, 2	6
14. DELETED					
15. Undervoltage-Reactor Coolant Pumps	2/bus	1/bus both busses	1/bus	1	6
16. Underfrequency-Reactor Coolant Pumps	3/bus	2 on same bus	2/bus	1	6
17. Turbine Trip					
a. Low Autostop Oil Pressure	3	2	2	1	7
b. Turbine Stop Valve Closure	4	4	4	1	7

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESF	2	1	2	1, 2	26
19. Reactor Coolant Pump Breaker Position Trip above P-7	1/breaker	2	1/breaker	1	9
20. Reactor Trip Breakers	2	1	2	1, 2	10, 12
	2	1	2	3*, 4*, 5*	11
21. Automatic Trip and Interlock Logic	2	1	2	1, 2	26
	2	1	2	3*, 4*, 5*	11
22. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7 P-10 Input	4	2	3	1	8#
P-13 Input	2	1	2	1	8#
c. Power Range Neutron Flux, P-8	4	2	3	1	8#
d. Power Range Neutron Flux, P-9	4	2	3	1	8#
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8#
f. Turbine Impulse Chamber Pressure, P-13 (Input to P-7)	2	1	2	1	8#
23. Seismic Trip	3 directions (x,y,z) in 3 locations	2/3 locations one direction	2/3 locations all directions	1, 2	13

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 9 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within the next 6 hours.

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

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

ACTION 12 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The Minimum Channels OPERABLE requirement is met, and

b. The inoperable channel is placed in the tripped conditions within 6 hours; however, the inoperable channel may be bypassed for up to 72 hours for surveillance testing per Specification 4.3.1.1 or for performing maintenance.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	N.A.
2.	Power Range, Neutron Flux	≤ 0.5 second*
3.	Power Range, Neutron Flux, High Positive Rate	N.A.
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5.	Intermediate Range, Neutron Flux	N.A.
6.	Source Range, Neutron Flux	≤ 0.5 second*
7.	Overtemperature ΔT	≤ 4 seconds*
8.	Overpower ΔT	N.A.
9.	Pressurizer Pressure-Low	≤ 2 seconds
10.	Pressurizer Pressure-High	≤ 2 seconds
11.	Pressurizer Water Level-High	N.A.
12.	Reactor Coolant Flow-Low	
	a. Single Loop (Above P-8)	≤ 1 second
	b. Two Loops (Above P-7 and below P-8)	≤ 1 second
13.	Steam Generator Water Level-Low-Low	≤ 2 seconds
14.	DELETED	
15.	Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
16.	Underfrequency-Reactor Coolant Pumps	≤ 0.6 second

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	Q	N.A.	N.A.	1, 2
14. DELETED						
15. Undervoltage-Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
16. Underfrequency-Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U(1, 9)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	S/U(1, 9)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A	1, 2
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	N.A.	1
20. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1

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
20. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
e. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
21. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 10)	N.A.	1, 2, 3*, 4*, 5*
22. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
23. Seismic Trip	N.A.	R	N.A.	R	R	1, 2
24. Reactor Trip Bypass Breaker	N.A	N.A.	N.A.	M(7,15),R(16)	N.A.	1,2,3*,4*,5*

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High	$\leq$ 3 psig	$\leq$ 3.5 psig
d. Pressurizer Pressure-Low	$\geq$ 1850 psig	$\geq$ 1840 psig
e. Differential Pressure Between Steam Lines-High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines-High	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
Coincident With Either		
1) T_{avg}-Low-Low, or	$\geq$ 543°F	$\geq$ 540.2°F
2) Steam Line Pressure-Low	$\geq$ 600 psig	$\geq$ 580 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 22 psig	≤ 24 psig
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
b. Phase "B" Isolation		
1) Manual	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure-High-High	≤ 22 psig	≤ 24 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation (Continued)		
c. Containment Ventilation Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Plant Vent Noble Gas Activity-High (RM-14A and 14B)[(a)]	Per the ODCP	
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)[(b)]	Per Specification 3.3.3.10	
4. Steam Line Isolation		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	$\leq$ 22 psig	$\leq$ 24 psig
d. Steam Flow in Two Steam Lines-High	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.

(a)The requirements for Plant Vent Noble Gas Activity-High (RM-14A and 14B) are not applicable following installation of RM-44A and 44B.

(b)The requirements for Containment Ventilation Exhaust Radiation-High (RM-44A and 44B) are applicable following installation of RM-44A and 44B.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
Coincident With Either		
1) T_{avg}-Low-Low, or	$\geq$ 543°F	$\geq$ 540.2°F
2) Steam Line Pressure-Low	$\geq$ 600 psig	$\geq$ 580 psig
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level-High-High	$\leq$ 67% of narrow range instrument span each steam generator.	$\leq$ 68% of narrow range instrument span each steam generator.
6. Auxiliary Feedwater		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-Low-Low	$\geq$ 7.2% of narrow range instrument span each steam generator.	$\geq$ 6.2% of narrow range instrument span each steam generator.
d. Undervoltage - RCP	$\geq$ 8050 volts	$\geq$ 7935 volts
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUES
7. Loss of Power (4.16 kV Emergency Bus Undervoltage)			
a. First Level			
1) Diesel Start		$\geq$ 0 volts with a $\leq$ 0.8 second time delay and $\geq$ 2583 volts with a $\leq$ 10 second time delay	$\geq$ 0 volts with a $\leq$ 0.8 second time delay and $\geq$ 2583 volts with a $\leq$ 10 second time delay
2) Initiation of Load Shed		One relay $\geq$ 0 volts with a $\leq$ 4 second time delay and $\geq$ 2583 volts with a $\leq$ 25 second time delay with one relay $\geq$ 2870 volts, instantaneous	One relay $\geq$ 0 volts with a $\leq$ 4 second time delay and $\geq$ 2583 volts with a $\leq$ 25 second time delay with one relay $\geq$ 2870 volts, instantaneous
b. Second Level			
1) Diesel Start		$\geq$ 3600 volts with a $\leq$ 10 second time delay	$\geq$ 3600 volts with a $\leq$ 10 second time delay
2) Initiation of Load Shed		$\geq$ 3600 volts with a $\leq$ 20 second time delay	$\geq$ 3600 volts with a $\leq$ 20 second time delay
8. Engineered Safety Features Actuation System Interlocks			
a. Pressurizer Pressure, P-11		$\leq$ 1915 psig	$\leq$ 1925 psig
b. Low-Low T_{avg}, P-12	increasing	543°F	$\leq$ 545.8°F
	decreasing	543°F	$\geq$ 540.2°F
c. Reactor Trip, P-4		N.A.	N.A.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
1) Feedwater Isolation	N.A.
2) Reactor Trip	N.A.
3) Phase "A" Isolation	N.A.
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	N.A.
6) Component Cooling Water	N.A.
7) Containment Fan Cooler Units	N.A.
8) Auxiliary Saltwater Pumps	N.A.
b. Phase "B" Isolation	
1) Containment Spray (Coincident with SI Signal)	N.A.
2) Containment Ventilation Isolation	N.A.
c. Phase "A" Isolation	
1) Containment Ventilation Isolation	N.A.
d. Steam Line Isolation	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(7)}/25^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 63^{(2)}$
3) Phase "A" Isolation	$\leq 18^{(1)}/28^{(3)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	$\leq 60^{(3)}$
6) Component Cooling Water	$\leq 38^{(1)}/48^{(3)}$
7) Containment Fan Cooler Units	$\leq 40^{(3)}$
8) Auxiliary Saltwater Pumps	$\leq 48^{(1)}/58^{(3)}$
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27^{(7)}/25^{(4)}/35^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 63^{(2)}$
3) Phase "A" Isolation	$\leq 18^{(1)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	$\leq 60^{(3)}$
6) Component Cooling Water	$\leq 48^{(3)}/38^{(1)}$
7) Containment Fan Cooler Units	$\leq 40^{(3)}$
8) Auxiliary Saltwater Pumps	$\leq 58^{(3)}/48^{(1)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 25^{(4)}/35^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 63^{(2)}$
3) Phase "A" Isolation	$\leq 18^{(1)}/28^{(3)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	$\leq 60^{(3)}$
6) Component Cooling Water	$\leq 38^{(1)}/48^{(3)}$
7) Containment Fan Cooler Units	$\leq 40^{(3)}$
8) Auxiliary Saltwater Pumps	$\leq 48^{(1)}/58^{(3)}$
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}-Low-Low	
a. Safety Injection (ECCS)	$\leq 25^{(4)}/35^{(5)}$
1) Reactor Trip	≤ 4
2) Feedwater Isolation	$\leq 65^{(2)}$
3) Phase "A" Isolation	$\leq 20^{(1)}/30^{(3)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	$\leq 60^{(3)}$
6) Component Cooling Water	$\leq 40^{(1)}/50^{(3)}$
7) Containment Fan Cooler Units	$\leq 40^{(3)}$
8) Auxiliary Saltwater Pumps	$\leq 50^{(1)}/60^{(3)}$
b. Steam Line Isolation	≤ 10
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 25^{(4)}/35^{(5)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 63^{(2)}$
3) Phase "A" Isolation	$\leq 18^{(1)}/28^{(3)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	$\leq 60^{(3)}$
6) Component Cooling Water	$\leq 38^{(1)}/48^{(3)}$
7) Containment Fan Cooler Units	$\leq 40^{(3)}$
8) Auxiliary Saltwater Pumps	$\leq 48^{(1)}/58^{(3)}$
b. Steam Line Isolation	≤ 8

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Containment Pressure-High-High	
a. Containment Spray	< 48.5(6)
b. Phase "B" Isolation	N.A.
c. Steam Line Isolation	≤ 7
8. Steam Generator Water Level-High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 66(2)
9. Steam Generator Water Level Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60(3)
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. RCP Bus Undervoltage	
Turbine-Driven Auxiliary Feedwater Pump	≤ 60
11. Plant Vent Noble Gas Activity-High(a)	
Containment Ventilation Isolation	≤ 11
12. Containment Ventilation Exhaust Radiation-High (b)	
Containment Ventilation Isolation	≤ 11

(a)The requirements for Plant Vent Noble Gas Activity-High are not applicable following installation of RM-44A and 44B.

(b)The requirements for Containment Ventilation Exhaust Radiation-High are applicable following installation of RM-44A and 44B.

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting delay not included because offsite power available.

(2) Feedwater System overall response time shall include verification of each individual Feedwater System valve closure time as shown below:

Valve	Closure Time (not including instrumentation delays)
FCV-438	$\leq$ 60 seconds
439	$\leq$ 60 seconds
440	$\leq$ 60 seconds
441	$\leq$ 60 seconds
510	$\leq$ 5 seconds
520	$\leq$ 5 seconds
530	$\leq$ 5 seconds
540	$\leq$ 5 seconds
1510	$\leq$ 5 seconds
1520	$\leq$ 5 seconds
1530	$\leq$ 5 seconds
1540	$\leq$ 5 seconds

(3) Diesel generator starting and loading delays included.

(4) Diesel generator starting delay not included because offsite power is available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps (where applicable). Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Offsite power is not available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(6) The maximum response time of 48.5 seconds is the time from when the containment pressure exceeds the High-High Setpoint until the spray pump is started and the discharge valve travels to the fully open position assuming off-site power is not available. The time of 48.5 seconds includes the 28-second maximum delay related to ESF loading sequence. Spray riser piping fill time is not included. The 80-second maximum spray delay time does not include the time from LOCA start to "P" signal.

(7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time limit includes opening of valves to establish SI flow path and attainment of discharge pressure for centrifugal charging pumps, SI, and RHR pumps (where applicable).

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place a residual heat removal train into operation.

SURVEILLANCE REQUIREMENTS

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

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

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. The provisions of Specification 3.0.4 may be suspended for up to 18 hours per valve for entry into and during operations in MODE 3 for the purpose of setting the pressurizer Code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

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

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 836 and 864 cubic feet of borated water,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

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

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

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

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected by sealing the breaker in the open position.

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

d. At least once per 18 months by 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 corrosion;

e. At least once per 18 months by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal.

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

      a) Centrifugal charging pump,

      b) Safety Injection pump, and

      c) Residual Heat Removal pump.

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 $\geq$ 2400 psid,

   2) Safety Injection pump $\geq$ 1455 psid, and

   3) Residual Heat Removal pump $\geq$ 165 psid.

g. By verifying the correct position of each electrical and/or 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) At least once per 18 months.

Charging Injection Throttle Valves	Safety Injection Throttle Valves
8810A	8822A
8810B	8822B
8810C	8822C
8810D	8822D

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

For Unit 1 Cycle 5

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 346 gpm, and

b) The total pump flow rate is less than or equal to 550 gpm.

2) For safety injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 463 gpm, and

b) The total pump flow rate is less than or equal to 650 gpm.

For Unit 1 Cycle 6 and after, and Unit 2 Cycle 5 and after:

1) For centrifugal charging pumps, with a single pump running:

a) The sum of injection line flow rates, excluding the highest flow rate, is greater than or equal to 299 gpm, and

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STEAM GENERATOR 10% ATMOSPHERIC DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Four steam generator 10% atmospheric dump valves (ADVs) with the associated block valves open and associated remote manual controls, including the backup air bottles, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one less than the required number of 10% ADVs OPERABLE, restore the inoperable steam generator 10% ADV to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two less than the required numbered of 10% ADVs OPERABLE, restore at least one of the inoperable steam generator 10% ADVs to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each steam generator 10% ADV, associated block valve and associated remote manual controls including the backup air bottles shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the backup air bottle for each steam generator 10% ADV has a pressure greater than or equal to 260 psig, and

b. At least once per 31 days by verifying that the steam generator 10% ADV block valves are open, and

c. At least once per 18 months by verifying that all steam generator 10% ADVs will operate using the remote manual controls and the backup air bottles.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

a. A K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2,000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

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

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

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in containment and the control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except for latching the control rod drive mechanism shaft to the rod cluster control assemblies and friction testing of individual control rods.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for the trip insertion 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 its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, 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 its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 1.

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

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 conditions assumed in the FSAR accident and transient analysis.

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

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC) was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections 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 MTC value -3.9×10^{-4} $\Delta k/k/°F$. The MTC value of -3.0×10^{-4} $\Delta k/k/°F$ 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 MTC value of -3.9×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of each 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. In addition, verification during startup testing at beginning of life hot zero power for each cycle validates that the MTC parameters are within the limits specified for all other power levels.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective 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.

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 BOL when borating from hot zero power to COLD SHUTDOWN and requires 14,042 gallons of 7,000 ppm borated water from the boric acid storage tanks or 65,784 gallons of 2300 ppm borated water from the refueling water storage tank.

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 change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,499 gallons of 7,000 ppm borated water from the boric acid storage tanks or 17,865 gallons of 2300 ppm borated water from the refueling water storage tank.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

ECCS SUBSYSTEMS (Continued)

The maximum flow Surveillance Requirement ensures that the minimum injection line resistance assumptions are met. These assumptions are used to calculate maximum flows to the RCS for safety analyses which are limited by maximum ECCS flow to the RCS.

The Surveillance Requirement for the maximum difference between the minimum and maximum individual injection line flows ensures that the minimum individual injection line resistance assumed for the spilling line following a LOCA is met.

The maximum total pump flow Surveillance Requirements ensure the pump runout limits of 560 gpm for the centrifugal charging pumps and 675 gpm for the safety injection pumps are met.

The safety analyses are performed assuming the miniflow recirculation lines for the ECCS subsystems associated with the centrifugal charging and safety injection pumps are open. The flow balancing test is, therefore, performed with these miniflow recirculation lines open.

Some of the flow from the centrifugal charging pumps will go to the RCP seals during ECCS operation. Therefore, the flow balance test is performed with a simulated flow from the centrifugal charging pumps to the RCP seals. The simulated flow rate is consistent with the actual RCP seal resistance and the resistance of the RCP seals assumed in the calculation of ECCS flows for the safety analyses.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core; (2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, spray additive tank, containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1); (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area greater than 3 ft^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO); and (4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter of February 4, 1992, Pacific Gas and Electric Company (or the licensee) submitted a request for changes to the Technical Specifications (TS). The proposed amendments would revise TS 2.2, 3/4.1.1, 3/4.1.2, 3/4.1.3, 3/4.2.5, 3/4.3.1, 3/4.3.2, 3/4.4.2, 3/4.5.1, 3/4.5.2, 3/4.5.4, 3/4.7.1, 3/4.9.1, and 3/4.10.1 by making administrative changes to remove cycle specific information that is no longer necessary and to correct Table 3.3-5 notations.

2.0 EVALUATION

The current DCPP Units 1 and 2 Technical Specifications (TS) contain information that is cycle specific. Because both units are in Cycle 5, references to previous cycles are outdated and thus no longer valid. Since the deletion of the outdated cycle specific information is an administrative change and would therefore have no effect on any plant systems or the safe operation of DCPP, the proposed changes are not considered to have any safety significance. The changes, would, however, clarify the TS by removing extraneous information.

In addition to removing cycle specific TS items the licensee proposes to correct table notations concerning Table 3.3-5, "Engineered Safety Features Response Times," which provides required response times for various initiating signals. The proposed changes to the Table 3.3-5 notations reflect current operating conditions at DCPP. The table notation changes were intended to be made as part of Amendments 51 and 50, but due to an administrative error by the licensee, the changes were omitted. The justification and safety analysis concerning the initiating signal and function response times for the containment pressure high, phase 'A' isolation component cooling water, and auxiliary salt water pump response times determined that these parameters are not subject to the time delays associated with the sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST). In addition, the initiating signal and function response time for containment pressure high, auxiliary feedwater, and containment fan cooler unit response times include the diesel generator starting and loading time delays. By including the diesel generator starting and loading time delays the DCPP TS will be consistent with current DCPP surveillance testing requirements.

Based on the above, the staff finds the proposed changes to remove outdated cycle specific information and correct Table 3.3-5 notations are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 13136). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: August 6, 1992