February 17, 1998

Mr. Gregory M. Rueger
Pacific Gas and Electric Company
NPG - Mail Code A10D
P. O. Box 770000
San Francisco, California 94177

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M97472) AND UNIT NO. 2 (TAC NO. M97473)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 122 to Facility Operating License No. DPR-80 and Amendment No. 120 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 December 9, 1996.

These amendments revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to change the surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) for the reactor trip system (RTS) and engineering safety features actuation systems (ESFAS) instrumentation channels, and make certain changes in trip setpoints and allowance values due to a setpoint methodology change in support of the calibration extensions. Channel operational tests (COTs) and trip actuating device operational tests (TADOTs) associated with these channels are also being extended. Revisions to the appropriate TS Bases are being revised to support the TS revisions.

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

Sincerely,

Original Signed By

Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosures: 1. Amendment No. 122 to DPR-80
2. Amendment No. 120 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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OFC	PDIV-2/PM	PDIV-2/LA	~~NRR:EELB~~	NRR:SRXB(A)	OGC
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DATE	8/26/97	8/25/97	~~8/ /97~~	10/30/97	1/6/97

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cc w/encls:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 122
License No. DPR-80

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated December 9, 1996, 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. 122 , are hereby incorporated in the license. Pacific Gas and 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 its date of issuance to be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven D. Bloom

Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 17, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-82

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

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated December 9, 1996, 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. 120 , are hereby incorporated in the license. Pacific Gas and 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 its date of issuance to be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven D. Bloom

Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: **February 17, 1998**

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 120 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 Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-4	2-4
2-5	2-5
2-6	2-6
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
B 2-3	B 2-3
B 2-4	B 2-4
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a
----------	B 3/4 3-1b

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

b. 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 requirements of Specification 3.3.1 until the channel is restored to OPERABLE status with its 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	≤ 26.2% of RATED THERMAL POWER
b. High Setpoint	≤ 109% of RATED THERMAL POWER	≤ 110.2% of RATED THERMAL POWER
3. Power Range, Neutron Flux High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.6% 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	≤ 5.6% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30.6% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.4 x 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	≥ 1947.5 psig
10. Pressurizer Pressure-High	≤ 2385 psig	≤ 2387.5 psig
11. Pressurizer Water Level-High	≤ 90% of instrument span	≤ 90.2% of instrument span
12. Reactor Coolant Flow-Low	≥ 90% of minimum measured flow** per loop	≥ 89.8% 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.

Unit 1 - Amendment No. 37,72,122
Unit 2 - Amendment No. 36,71,120

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	≥ 7.0% of narrow range instrument span-each steam generator
Coincident with:		
a. RCS Loop ΔT Equivalent to Power ≤50% RTP	RCS Loop ΔT variable input ≤50% RTP	RCS Loop ΔT variable input ≤50.7% RTP
With a time delay (TD)	≤TD (Note 5)	≤(1.01)TD (Note 5)
Or		
b. RCS Loop ΔT Equivalent to Power >50% RTP	RCS Loop ΔT variable input >50% RTP	RCS Loop ΔT variable input >50.7% RTP
With no time delay	TD = 0	TD = 0
14. DELETED		
15. Undervoltage-Reactor Coolant Pumps	≥ 8050 volts-each bus	≥ 7877 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	$\geq 8 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	$\geq$ 8.8%, $\leq$ 11.2% of RATED THERMAL POWER
2) P-13 Input	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 10.2% RTP Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	$\leq$ 35% of RATED THERMAL POWER	$\leq$ 36.2% of RATED THERMAL POWER
d. Power Range Neutron Flux, P-9	$\leq$ 50% of RATED THERMAL POWER	$\leq$ 51.2% of RATED THERMAL POWER
e. Power Range Neutron Flux, P-10	10% of RATED THERMAL POWER	$\geq$ 8.8%, $\leq$ 11.2% of RATED THERMAL POWER
f. Turbine Impulse Chamber Pressure, P-13	$\leq$ 10% RTP Turbine Impulse Pressure Equivalent	$\leq$ 10.2% RTP Turbine Impulse Pressure Equivalent
23. Seismic Trip	$\leq$ 0.35 g	$\leq$ 0.43 g

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = Lead-lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for ΔT, τ_4 = 0 seconds, τ_5 = 0 seconds

ΔT_o = Loop specific indicated ΔT at RATED THERMAL POWER

K_1 = 1.2

K_2 = 0.0182/°F

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

τ_1, τ_2 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_1 = 30 seconds, τ_2 = 4 seconds

T = Average temperature, °F

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: (continued)

T'	= Nominal loop specific indicated T_{avg} at RATED THERMAL POWER
K_3	= 0.000831/psig
P	= Pressurizer pressure, psig
P'	= 2235 psig (Nominal RCS operating pressure)
S	= Laplace transform operator, s^{-1}

and $f_1(\Delta I)$ is a function of the indicated difference between top and between 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 +7%, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(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 +7%, the ΔT Trip Setpoint shall be automatically reduced by 2.38% 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 0.46% ΔT span for hot leg or cold leg temperature inputs, 0.14% ΔT span for pressurizer pressure input, or 0.19% ΔT span for ΔI inputs.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 3: Overpower ΔT

$$\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \left\{ K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 [T - T''] - f_2(\Delta I) \right\}$$

Where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = Lead-lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for ΔT, τ_4 = 0 seconds, τ_5 = 0 seconds

ΔT_o = Loop Specific Indicated ΔT at RATED THERMAL POWER

K_4 = 1.072

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

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

τ_3 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_3 = 10 secs.

K_6 = 0.00145/°F for T > T", and 0 for T ≤ T"

T = Average temperature, °F

T" = Nominal loop specific indicated T_{avg} at RATED THERMAL POWER

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

$f_2(\Delta I)$ = 0 for all ΔI

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 0.46% ΔT span for hot leg or cold leg temperature inputs.

NOTE 5: Steam Generator Water Level Low-Low Trip Time Delay

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), $P \leq 50\%$ RTP

TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

B1 = -0.007128

B2 = +0.8099

B3 = -31.40

B4 = +464.1

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Allowable Values are considered the Limiting Safety System Settings (LSSS) as identified in 10 CFR 50.36. The LSSS settings have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore, providing protection system functional diversity.

The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as left" setpoint is within the band allowed for calibration tolerance. There is a band allowed for calibration tolerance only for those setpoints which use analog hardware. For example, the Power Range, Neutron Flux High setpoint is properly adjusted when it is set at 109% ± 0.3% (0.25% of 120% power span). The calibration tolerance, after appropriate conversion, should correspond to the rack comparator setting accuracy defined in the latest setpoint study. The setpoints which use digital hardware are set at the nominal value in the system.

Trip setpoints may be administratively redefined in the conservative direction for several reasons including startup, testing, process error accountability, or even a conservative response for equipment malfunction or inoperability. Some trip functions have historically been redefined at the beginning of each cycle for purposes of startup testing, e.g., Power Range Neutron Flux High and Overtemperature ΔT. Calibration to within the defined calibration tolerance of an administratively redefined, conservative Trip Setpoint is acceptable. Redefinition at full power conditions for these functions is expected and acceptable.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with a trip set less conservative than its Trip Setpoint, but within its specified Allowable Value, is acceptable. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Since there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift in excess of the allowance that is more than occasional may be indicative of more serious problems and warrants further investigation.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

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

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

Power Range, Neutron Flux, High Rates

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

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBRs will be greater than or equal to the DNBR limits.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Trip System.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity, e.g., no fuel pellet cracking or melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
17. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve	N.A.
18. Safety Injection Input from ESF	N.A.
19. Reactor Coolant Pump Breaker Position Trip	N.A.
20. Reactor Trip Breakers	N.A.
21. Automatic Trip and Interlock Logic	N.A.
22. Reactor Trip System Interlocks	N.A.
23. Seismic Trip	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	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.	R24(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R24(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R24(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R24(4)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R24(4)	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R24(4, 5)	S/U(1)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R24(4, 5)	S/U(1), Q(8)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R24	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R24	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low	S	R24	Q	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R24	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High	S	R24	Q	N.A.	N.A.	1
12. Reactor Coolant Flow-Low	S	R24	Q	N.A.	N.A.	1

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low						
a. Steam Generator Water Level-Low-Low	S	R24	Q	N.A.	N.A.	1, 2
b. RCS Loop ΔT Equivalent to Power	N.A.	R24	Q	N.A.	N.A.	1, 2
14. DELETED						
15. Undervoltage-Reactor Coolant Pumps	N.A.	R24	N.A.	Q	N.A.	1
16. Underfrequency-Reactor Coolant Pumps	N.A.	R24	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.	R24	N.A	1, 2
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R24	N.A.	1
20. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R24(4)	R24	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R24(4)	R24	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R24(4)	R24	N.A.	N.A.	1

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	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.	R24(4)	R24	N.A.	N.A.	1
e. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R24(4)	R24	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R24	R24	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.	R24	N.A.	R24	M(7)	1, 2
24. Reactor Trip Bypass Breaker	N.A	N.A.	N.A.	M(7,15),R24(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	≤ 3 psig	≤ 3.12 psig
d. Pressurizer Pressure-Low	≥ 1850 psig	≥ 1847.5 psig
e. DELETED		
f. Steam Line Pressure-Low	≥ 600 psig (Note 1)	≥ 597.6 psig (Note 1)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Spray (coincident with SI signal)		
a. Manual Initiation	N.A.	N.A
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A
c. Containment Pressure-High-High	$\leq$ 22 psig	$\leq$ 22.12 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	$\leq$ 22 psig	$\leq$ 22.12 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) Deleted		
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)	Per the ODCP	
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	≤ 22 psig	≤ 22.12 psig
d. Steam Line Pressure-Low	≥ 600 psig (Note 1)	≥ 597.6 psig (Note 1)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
e. Negative Steam Line Pressure Rate-High	≤ 100 psi (Note 3)	≤ 102.4 psi (Note 3)
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level-High-High	≤ 75% of narrow range instrument span each steam generator.	≤ 75.2% 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	≥ 7.2% of narrow range instrument span each steam generator.	≥ 7.0% of narrow range instrument span each steam generator.
Coincident with:		
1) RCS Loop ΔT Equivalent to Power ≤ 50% RTP	RCS Loop ΔT variable input ≤ 50% RTP	RCS Loop ΔT variable input ≤ 50.7% RTP
With a time delay (TD)	≤ TD (Note 2)	≤ (1.01)TD (Note 2)
Or		
2) RCS Loop ΔT Equivalent to Power > 50% RTP	RCS Loop ΔT variable input >50% RTP	RCS Loop ΔT variable input >50.7% RTP
With no time delay	TD = 0	TD = 0
d. Undervoltage - RCP	≥ 8050 volts	≥ 7877 volts
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

TABLE 3.3-4 (Continued)

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	≥ 0 volts with a ≤ 0.8 second time delay and ≥ 2583 volts with a ≤ 10 second time delay	≥ 0 volts with a ≤ 0.8 second time delay and ≥ 2583 volts with a ≤ 10 second time delay
2) Initiation of Load Shed	One relay ≥ 0 volts with a ≤ 4 second time delay and ≥ 2583 volts with a ≤ 25 second time delay with one relay ≥ 2870 volts, instantaneous	One relay ≥ 0 volts with a ≤ 4 second time delay and ≥ 2583 volts with a ≤ 25 second time delay with one relay ≥ 2870 volts, instantaneous
b. Second Level		
1) Diesel Start	≥ 3785 volts with a ≤ 10 second time delay	≥ 3785 volts with a ≤ 10 second time delay
2) Initiation of Load Shed	≥ 3785 volts with a ≤ 20 second time delay	≥ 3785 volts with a ≤ 20 second time delay
8. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	≤ 1915 psig	≤ 1917.5 psig
b. DELETED		
c. Reactor Trip, P-4	N.A.	N.A.

NOTE 1: Time constants utilized in the lead-lag compensator for Steam Pressure - Low are τ_1 = 50 seconds and τ_2 = 5 seconds.

NOTE 2: Steam Generator Water Level Low-Low Trip Time Delay

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), P ≤ 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low (in seconds)

B1 = -0.007128
B2 = +0.8099
B3 = -31.40
B4 = +464.1

NOTE 3: Time constants utilized in the rate-lag compensator for Negative Steam Line Pressure Rate - High are τ_3 = 50 seconds and τ_4 = 50 seconds.

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	≤ 63
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	≤ 63
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 NOTATIONS

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

(2) Notation deleted.

(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).

(8) Does not include Trip Time Delays.. Response times include the transmitters, Eagle-21 Process Protection cabinets, Solid State Protection System cabinets and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 50% RTP.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
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.	N.A.	R24	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
c. Containment Pressure-High	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. DELETED								
f. Steam Line Pressure-Low	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray (coincident with SI signal)								
a. Manual Initiation	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
c. Containment Pressure-High-High	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
3) Safety Injection		See Item 1. above for all Safety Injection Surveillance Requirements.						
b. Phase "B" Isolation								
1) Manual	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
3) Containment Pressure-High-High	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Containment Ventilation Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
2) Deleted								
3) Safety Injection		See Item 1. above for all Safety Injection Surveillance Requirements.						
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual	N.A.	N.A.	N.A.	R24	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)	R	1, 2, 3
c. Containment Pressure-High-High	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Negative Steam Line Pressure Rate-High	S	R24	Q	N.A.	N.A.	N.A.	N.A.	3(3)
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)	R	1, 2
b. Steam Generator Water Level-High-High	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2
6. Auxiliary Feedwater								
a. Manual	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)	R	1, 2, 3
c. Steam Generator Water Level-Low-Low								
1) Steam Generator Water Level-Low-Low	S	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3(5)
2) RCS Loop ΔT Equivalent to Power	N.A.	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
d. Undervoltage - RCP	N.A.	R24	N.A.	R24	N.A.	N.A.	N.A.	1
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
7. Loss of Power								
a. 4.16 kV Emergency Bus Level 1	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV Emergency Bus Level 2	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
8. Engineered Safety Feature Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R24	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Deleted								
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	1, 2, 3

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) For the Containment Ventilation Exhaust Radiation-High monitor only, a CHANNEL FUNCTIONAL TEST shall be performed at least once every 31 days.
(3) Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.
(4) Deleted.
(5) For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level-Low-Low channel must be less than or equal to 464.1 seconds.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and Engineered Safety Features Actuation System (ESFAS) instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out-of-service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System.

The Process Protection System is designed to permit any one channel to be tested and maintained at power in a bypassed mode. If a channel has been bypassed for any purpose, the bypass is continuously indicated in the control room as required by applicable codes and standards. As an alternative to testing in the bypass mode, testing in the trip mode is also possible and permitted.

The ESFAS senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the ESFAS to mitigate the consequences of a steam line break or loss of coolant accident: (1) safety injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valve position, (10) containment fan cooler units start, and (11) component cooling water pumps start and automatic valves position.

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

The ESFAS Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the trips are set for each functional unit. The Allowable Values are considred to be the Limiting Safety System Settings (LSSS) as identified in 10 CFR 50.36 and have been selected to mitigate the consequences of accidents. If the functional unit is based on analog hardware, the setpoint is considered to be adjusted consistent with the nominal value when the "as left" setpoint is within the band allowed for calibration tolerance. The calibration tolerance, after appropriate conversion, should correspond to the rack comparator setting accuracy defined in the latest setpoint study. For all setpoints in digital hardware, the setpoints are set at the nominal values.

The ESFAS Trip Setpoints may be administratively redefined in the conservative direction for several reasons including startup, testing, process error accountability, or even a conservative response for equipment malfunction or inoperability. ESFAS functions are not historically redefined at the beginning of each cycle for purposes of startup or testing as several Reactor Trip functions are. However, calibration to within the defined calibration tolerance of an administratively redefined, conservative Trip Setpoint is acceptable. Redefinition at full power conditions for these functions is expected and acceptable.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint, but within the Allowable Value, is acceptable. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Since there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift in excess of the allowance that is more than occasional may be indicative of more serious problems and warrants further investigation.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channel. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

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

ESF response times specified in Table 3.3-5, which include sequential operation of the RWST and VCT valves (Table Notations 4 and 5), are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction isolation valves are closed following opening of the RWST charging pump suction isolation valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Table Notation 7), the values specified are based on the LOCA analyses. The LOCA analyses takes credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

For slave relays in the ESF actuation system circuit that are Potter & Brumfield type MDR relays, the SLAVE RELAY TEST is performed on a refueling frequency. The test frequency is based on relay reliability assessments presented in WCAP-13878, "Reliability Assessment of Potter and Brumfield MDR Series Relays," WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," and WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relays." These reliability assessments are relay specific and apply only to Potter and Brumfield MDR series relays. Note that for normally energized applications, the relays may have to be replaced periodically in accordance with the guidance given in WCAP-13878 for MDR relays.

Undervoltage protection will generate a loss of power diesel generator start in the event a loss of voltage or degraded voltage condition occurs. The diesel generators provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. The first level undervoltage relays (FLURs) detect the loss of bus voltage (less than 69% bus voltage). The second level undervoltage relays (SLURS) provide a second level of undervoltage protection which protects all Class 1E loads from short or long term degradation in the offsite power system. The SLUR allowable value is the minimum steady state voltage needed on the 4160 volt vital bus to ensure adequate voltage is available for safety related equipment at the 4160 volt, 480 volt, and 120 volt levels.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 120 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 application dated December 9, 1996, Pacific Gas and Electric Company (or the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. DPR-80 and DPR-82) for the Diablo Canyon Nuclear Power Plant, Units 1 and 2. The proposed changes revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to;

a. Revise "Allowable Values" (AV) for the instrumentation of the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS) functions, to implement results of new setpoint calculations performed to support extension of surveillance test intervals (STIs) of these functions from 18 to 24 months not to exceed 30 months, and revise the setpoint for the pressurizer water level-high reactor trip function to provide adequate margin for the instrument loop uncertainties.

b. Extend STIs of TS surveillances including channel operational tests (COTs) and trip actuating device operational tests (TADOTs) of instrument channels associated with several RTS and the ESFAS functions, from at least once every 18 months to at least once per refueling interval of nominally 24 months, not to exceed 30 months.

c. Revise affected sections of the BASES to reflect changes in AVs and STIs of RTS and ESFAS Functional Units.

d. Incorporate editorial changes to clarify selected footnotes of TS Tables.

2.0 BACKGROUND

The licensee recently conducted a feasibility study for increasing the fuel cycles length from the current 18 months to 24 months for both units of DCPP. The results of this study indicated that a 24-month fuel cycle is not only feasible but also is beneficial because of fewer refuelings, improved outage scheduling and reduced personnel dose. Therefore, the licensee has decided to implement the extended 24-month fuel cycles at both units of the DCPP, extending the current refueling interval from 18 months to 24 months.

Current DCPP TS requires that surveillance tests for some functional units must be performed at least once per refueling interval. Therefore, STIs for these functional units have been identified by a notation "R" in an appropriate column of the current TS. With the extended fuel cycle, STI for these functional units will be 24 months. Therefore, for the 24-month (new refueling interval) STI, a new notation "R24" will be used. The licensee has decided to retain the current 18-month STI for some functional units. In their submittal, the licensee indicated that they would retain the existing notation "R" to indicate the 18-month STI. This will allow clear differentiation between 24-month and 18-month STIs.

3.0 EVALUATION

3.1 Allowable Values and Setpoints

Proposed TS change: Revise allowable value (AV) and/or setpoint (SP) for functional units of the following TS Tables as noted.

TS 2.2.1, Table 2.2-1, Reactor Trip System Instrumentation Trip

1. Functional Unit 2.a - Power Range, Neutron Flux, Low Setpoint, revise AV from ≤ 27.1% of RATED THERMAL POWER to ≤ 26.2% of RATED THERMAL POWER.

2. Functional Unit 2.b - Power Range, Neutron Flux, High Setpoint, revise AV from ≤ 111.1% of RATED THERMAL POWER to ≤ 110.2% of RATED THERMAL POWER.

3. Functional Unit 3 - Power Range, Neutron Flux High Positive Rate, revise AV from ≤ 6.5% of RATED THERMAL POWER to ≤ 5.6% of RATED THERMAL POWER.

4. Functional Unit 4 - Power Range, Neutron Flux High Negative Rate, revise AV from ≤ 6.5% of RATED THERMAL POWER to ≤ 5.6% of RATED THERMAL POWER.

5. Functional Unit 5 - Intermediate Range, Neutron Flux, revise AV from ≤ 30.9% of RATED THERMAL POWER to ≤30.6% of RATED THERMAL POWER.

6. Functional Unit 9 - Pressurizer Pressure-Low, revise AV from ≥ 1944.4 psig to ≥ 1947.5 psig.

7. Functional Unit 10 - Pressurizer Pressure-High, revise AV from ≤ 2390.6 psig to ≤ 2387.5 psig.

8. Functional Unit 11 - Pressurizer Water Level-High, revise AV from ≤ 92.5% to ≤ 90.2% of instrument span and revise trip setpoint from ≤ 92% to ≤ 90% of instrument span.

9. Functional Unit 12 - Reactor Coolant Flow-Low, revise AV from $\geq$ 89.7% to $\geq$ 89.8% of measured flow per loop.

10. Functional Unit 13 - Steam Generator Water Level Low-Low, revise AV from $\geq$ 6.8% to $\geq$ 7.0% of narrow range instrument span each-steam generator.

11. Functional Unit 13.a and 13.b - Steam Generator Water Level, Low-Low, RCS Loop Delta T equivalent to power > 50% RTP, revise AV from > 51.5% to > 50.7% RTP.

12. Functional Unit 15 - Undervoltage-Reactor Coolant Pumps, revise AV from $\geq$ 7730 volts to $\geq$ 7877 volts.

13. Functional Unit 22.a - Reactor Trip System Interlocks, Intermediate Range Neutron Flux, P-6, revise AV from $\geq 6\times10^{-11}$ to $\geq 8\times10^{-11}$ amps.

14. Functional Unit 22.b.1) - Reactor Trip System Interlocks, Low Power Reactor Trips Block, P-7, P-10 Input, revise AV from $\geq$ 7.9% $\leq$ 12.1% to $\geq$ 8.8% $\leq$ 11.2% of RATED THERMAL POWER.

15. Functional Unit 22.b.2) - Reactor Trip System Interlocks, Low Power Reactor Trips Block, P-7, P-13 Input, revise AV from $\leq$ 12.1% to $\leq$10.2% RTP turbine impulse pressure equivalent.

16. Functional Unit 22.c - Reactor Trip System Interlocks, Power Range Neutron Flux, P-8, revise AV from $\leq$ 37.1% to $\leq$ 36.2% of RATED THERMAL POWER.

17. Functional Unit 22.d - Reactor Trip System Interlocks, Power Range Neutron Flux, P-9, revise AV from $\leq$ 52.1% to $\leq$ 51.2% of RATED THERMAL POWER.

18. Functional Unit 22.e - Reactor Trip System Interlocks, Power Range Neutron Flux, P-10, revise AV from $\geq$ 7.9% $\leq$ 12.1% to $\geq$ 8.8% $\leq$ 11.2% of RATED THERMAL POWER.

19. Functional Unit 22.f - Reactor Trip System Interlocks, Turbine Impulse Chamber Pressure, P-13, revise AV from $\leq$ 12.1% to $\leq$ 10.2% RTP turbine impulse pressure equivalent.

20. Functional Unit 23 - Seismic Trip, revise AV from $\leq$ 0.40g to $\leq$ 0.43g.

21. Table 2.2-1, Over Temperature Delta T Note 2 revision. Replace words "1.0% Delta T span" with words, "0.46% Delta T span for hot leg or cold leg temperature inputs, 0.14% Delta T span for pressurizer pressure input, or 0.19% Delta T span for Delta I inputs".

22. Table 2.2-1, Overpower Delta T Note 4 revision. Replace words, "1.0%" with "0.46%", and add words "for hot leg or cold leg temperature inputs", after the word "span".

TS 3/4.3.2, Table 3.3-4, Engineered Safety Features Actuation System Instrumentation

1. Functional Unit 1.c - Safety Injection, Containment Pressure - High, revise AV from ≤ 3.3 psig to ≤ 3.12 psig.

2. Functional Unit 1.d - Safety Injection, Pressurizer Pressure - Low, revise AV from ≥ 1844.4 psig to ≥ 1847.5 psig.

3. Functional Unit 1.f - Safety Injection, Steam Line Pressure - Low, revise AV from ≥ 594.6 psig to ≥ 597.6 psig.

4. Functional Unit 2.c - Containment Spray, Containment Pressure - High-High, revise AV from ≤ 22.3 psig to ≤ 22.12 psig.

5. Functional Unit 3.b.3) - Containment Isolation, Phase B Isolation, Containment Pressure - High-High, revise AV from ≤ 22.3 psig to ≤ 22.12 psig.

6. Functional Unit 4.c - Steam Line Isolation, Containment Pressure - High-High, revise AV from ≤ 22.3 psig to ≤ 22.12 psig.

7. Functional Unit 4.d - Steam Line Isolation, Steam Line Pressure Low, revise AV from ≥ 594.6 psig to ≥ 597.6 psig.

8. Functional Unit 4.e - Steam Line Isolation - Negative Steam Line Pressure Rate - High, revise AV from ≤ 105.4 psi to ≤ 102.4 psi.

9. Functional Unit 5.b - Turbine Trip and Feedwater Isolation, Steam Generator Water Level - High-High, revise AV from ≤ 75.5% to ≤ 75.2%.

10. Functional Unit 6.c - Auxiliary Feedwater, Steam Generator Water Level - Low-Low, Initiation on Steam Generator Water Level - Low-Low, revise AV from ≥ 6.8% to ≥ 7.0%.

11. Functional Unit 6.c.1) and 2) - Auxiliary Feedwater, Steam Generator Water Level - Low-Low, RCS Loop Delta T equivalent to Power, revise AV from ≤ 51.5% RTP to ≤ 50.7% RTP.

12. Functional Unit 6.d - Auxiliary Feedwater, Undervoltage - RCP, revise AV from ≥ 7730 volts to ≥ 7877 volts.

13. Functional Unit 8.a - ESFAS Interlocks, Pressurizer Pressure, revise AV from ≤ 1920.6 psig to ≤ 1917.5 psig.

3.1.2 <u>Justification for the Change</u>

In their submittal, the licensee stated that their calculations to revise the AV and SP of functional units to support STI extensions were based on methodologies of WCAP-11082, Revision 5, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle

Evaluation," and WCAP-14646, Revision 0, "Instrumentation Calibration and Drift Evaluation Process for Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle." The proposed TS format for the trip setpoints and allowable values is consistent with that currently used at DCPP, and is also consistent with the two column format used in NUREG-1431, Revision 1, "Improved Standard Technical Specifications - Westinghouse Plants."

In their submittal, the licensee stated that evaluations in WCAP-11082 were based on a methodology responsive to the guidance in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle" dated April 2, 1991 which states that: that uncertainty calculations should be performed in a manner which results in drift values at a 95% probability and a 95% confidence level. The licensee stated that for the limiting safety system setting (LSSS) of each function, a trip setpoint and its associated allowable value have been established consistent with a 95/95 confidence level. The trip setpoint for each LSSS was selected such that adequate protection is provided when all uncertainties associated with the process, sensor, and rack are taken into account. During surveillance, the as-found value of the SP is compared to its AV and if the SP was found outside of its AV, the SP is reset within its as-left tolerance; if the SP could not be set within its as-left tolerance, then the channel is declared inoperable. Therefore, the trip SP and its AV together ensure that safety limits will not be exceeded. New values for most of the AVs have been proposed to support the calibration interval extension. SPs were not revised because of the adequacy of existing margins between the AVs and the SPs.

In their submittal, the licensee stated that uncertainties for several RTS channels were not recalculated using the 24-month fuel cycle algorithm. These channels are (1) the power range neutron flux high and low setpoints; (2) the power range neutron flux high positive and high negative rate setpoints; (3) the intermediate range neutron flux setpoints; and (4) the source range neutron flux setpoints. The channel statistical allowance (CSA) equation (algorithm) used for combining the uncertainty components for these channels is the same one used in Revision 2 of WCAP-11082, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Stations, Eagle 21 Version." supporting the current DCPP TS setpoints for these channels. The licensee stated that the CSA equation results in more conservative treatment of uncertainties than would be obtained if these channels were reevaluated with the equation from WCAP-11082, Rev. 5, for the 24-month fuel cycle. The staff agrees with the licensee and finds the use of WCAP-11082, Rev. 2 for uncertainties to be acceptable.

For pressurizer level, item numbers 8 and 36 of the submittal, the licensee stated that the results of the channel statistical calculations for pressurizer level show that the total channel uncertainty was found to be 9.12 percent of span, which exceeds that which can be supported by the current TS setpoint for the reactor trip function. There is no explicit safety analysis limit identified for this function. However, in order to assure that a steam bubble always exists within the pressurizer for pressurizer pressure control, the total channel uncertainty must be accommodated between the trip setpoint and the top of the span. A pressurizer level setpoint change is proposed in order to support an extended surveillance of up to 30 months. The allowable value is also proposed to be changed to be consistent with the new setpoint. These new values will be established in accordance with the above methodology which is acceptable to the staff.

For the RCS loss of flow-low, proposed revision item numbers 9 and 37 of the submittal, the licensee stated that the 30-month CSA was larger than that previously calculated, and it approached the allowance between the TS setpoint and the safety analysis limit. Westinghouse evaluated the safety analysis limit to determine whether it could be reduced so as to support the existing setpoint of greater than or equal to 90 percent on minimum measured flow (MMF) per loop. The existing DCPP safety analysis is based on an assumed RCS loss of flow-low limit of 87 percent of MMF. RCS loss of flow-low is credited in two analyzed accidents, the partial loss of forced reactor coolant flow and the single RCP locked rotor. The Westinghouse evaluation concluded that the limit can be lowered to 85 percent of MMF, with no impact on the DCPP safety analysis, and that the existing conclusions in the DCPP FSAR Update remain valid. The lower safety limit of 85 percent of MMF provides sufficient allowance for the RCS loss of flow-low setpoint considering the new CSA (WCAP-11082, Rev. 5), therefore no setpoint change is required. Due to the change in setpoint methodology, there is a change in the AV to greater than or equal to 89.8 percent of MMF in order to support an extended surveillance of up to 30 months. The AV change is in the more restrictive direction and is consistent with the WCAP-11082, Revision 5, methodology.

For the Undervoltage - RCP, proposed revision item numbers 11, 40, 59 and 72 of the submittal, the licensee stated that following replacement of the original Westinghouse undervoltage relays with new Basler Model BE1-27 relays, they had a discussion with the vendor since the instruction manual does not provide a specific drift specification. The vendor confirmed that the expected relay drop-out and pick-up setting drift is less than 1.6 volts between calibration, therefore, a ± 1.6 volt drift allowance was used for 30 months and will be verified based on future performance data. The licensee's evaluation to determine other channel uncertainties was based on the currently installed hardware as defined by their calibration procedures. The total channel uncertainty for the RCP undervoltage channel was calculated to be 2.56 volts. The safety analyses do not assume an explicit value, therefore a safety analysis limit does not apply and there are no TS setpoint changes or safety analysis limit changes. For the undervoltage function, an AV change to greater than or equal to 7877 volts was proposed, which is more restrictive than the existing value and is consistent with WCAP-11082, Revision 5, methodology. Following the relay replacement, the TADOT and the calibration procedure were merged and are currently being performed quarterly. Based on the results of the recent tests on the replacement relays, a review of the past tests on the original relays indicated no problems, along with the intention to continue performing the TADOT's quarterly, the relays are expected to perform satisfactorily over a 24-month fuel cycle.

For the Underfrequency - RCP, proposed revision item number 41 of the submittal, the licensee stated that the evaluation on the Basler BE1-81, underfrequency relays resulted in a ± 0.04 Hz drift allowance for a 30-month calibration interval. The licensee's evaluation to determine other channel uncertainties was based on the currently installed hardware as defined by their calibration procedures. The total channel uncertainty for the RCP underfrequency reactor trip was calculated to be 0.091 Hz. The margin to the safety analysis limit is 0.009 Hz for the trip setpoint. Therefore, no changes in safety analysis limits, setpoints or allowable values are necessary.

The staff has reviewed the licensee's methodology for calculation of the AVs and SPs as described in WCAP-11082, Revision 5 and finds it consistent with the guidance in Regulatory Guide 1.105 and, therefore, acceptable.

3.2 Surveillance Test Interval

Proposed TS change: TS 3/4.3.1, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements and TS 3/4.3.2, Table 4.3.2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements. Revise surveillance test intervals as follows:

TS 3/4.3.1, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements. For the following functional units, revise channel calibration frequency from "R" to "R24".

1. Functional Unit 2.a - Power Range, Neutron Flux, High Setpoint

2. Functional Unit 2.b - Power Range, Neutron Flux, Low Setpoint

3. Functional Unit 3 - Power Range, Neutron Flux, High Positive Rate

4. Functional Unit 4 - Power Range, Neutron Flux, High Negative Rate

5. Functional Unit 5 - Intermediate Range, Neutron Flux

6. Functional Unit 6 - Source Range, Neutron Flux

7. Functional Unit 7 - Over Temperature Delta T (OT Delta T)

8. Functional Unit 8 - Overpower Delta T (OP Delta T)

9. Functional Unit 9 - Pressurizer Pressure, Low

10. Functional Unit 10 - Pressurizer Pressure, High

11. Functional Unit 11 - Pressurizer Water Level, High

12. Functional Unit 12 - Reactor Coolant Flow, Low

13. Functional Unit 13.a - Steam Generator Water Level, Low-Low Reactor Trip

14. Functional Unit 13.b - Steam Generator Water Level, Low-Low, RCS Loop Delta T input

15. Functional Unit 15 - Undervoltage-Reactor Coolant Pumps

16. Functional Unit 16 - Underfrequency-Reactor Coolant Pumps

TS 3/4.3.2, Table 4.3.2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements. For the following functional units, revise channel calibration frequency from "R" to "R24".

1. Functional Unit 1.c - Safety Injection, Containment Pressure - High

2. Functional Unit 1.d - Safety Injection, Pressurizer Pressure - Low

3. Functional Unit 1.f - Safety Injection, Steam Line Pressure - Low

4. Functional Unit 2.c - Containment Spray, Containment Pressure - High-High

5. Functional Unit 3.b.3) - Containment Isolation, Phase B Isolation, Containment Pressure - High-High

6. Functional Unit 4.c - Steam Line Isolation, Containment Pressure - High-High

7. Functional Unit 4.d - Steam Line Isolation, Steam Line Pressure - Low

8. Functional Unit 4.e - Steam Line Isolation, Negative Steam Line Pressure Rate - High

9. Functional Unit 5.b - Turbine Trip and Feedwater Isolation, Steam Generator Water Level - High-High

10. Functional Unit 6.c.1) -Auxiliary Feedwater, Steam Generator Water Level - Low-Low, Initiation on Steam Generator Water Level-Low-Low

11. Functional Unit 6.c.2) - Auxiliary Feedwater, Steam Generator Water Level- Low-Low, RCS Loop Delta T Input

12. Functional Unit 8.a. - ESFAS Interlocks, Pressurizer Pressure, P-11

TS 3/4.3.1, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements. For the following Functional Units, revise Channel Calibration and Channel Operational Test frequency from "R" to "R24".

1. Functional Unit 20.a - Reactor Trip System Interlocks, Intermediate Range Neutron Flux, P-6

2. Functional Unit 20.b - Reactor Trip System Interlocks, Low Power Reactor Trips Block, P-7

3. Functional Unit 20.c - Reactor Trip System Interlocks, Power Range Neutron Flux, P-8

4. Functional Unit 20.d - Reactor Trip System Interlocks, Power Range Neutron Flux, P-9

5. Functional Unit 20.e - Reactor Trip System Interlocks, Low Setpoint Power Range Neutron Flux, P-10

6. Functional Unit 20.f - Reactor Trip System Interlocks, Turbine Impulse Chamber Pressure, P-13

TS 3/4.3.1, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements. For Functional Unit 23 - Seismic Trip, revise channel calibration and trip actuating device operational test frequency from "R" to "R24" and actuation logic test frequency from "R" to "M(7)" or at least once per month with footnote 7 applying.

TS 3/4.3.2, Table 4.3.2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements. For Functional Unit 6.d. - Auxiliary Feedwater, Undervoltage - RCP, revise channel calibration frequency and trip actuating device operational test frequency from "R" to "R24.

<u>Justification for the Change</u>

In its submittal, the licensee stated that the proposed surveillance interval modifications are based on guidance provided in GL 91-04, regarding how licensees should evaluate the effects of a 24-month surveillance interval in order to confirm that such an extension has an insignificant impact on plant safety. The licensee has performed analyses of all affected instrument loops in order to establish the effect of a 30 month (24 months + 25% allowable tolerance) calibration frequency.

Using the in-house procedures and WCAP-14646, Revision 1, "Instrument Calibration and Drift Evaluation Process for Diablo Canyon Units 1 and 2, 24-month Fuel Cycle Evaluation," the analyses were performed to verify that the surveillance interval extensions have an insignificant effect on plant safety and would not invalidate any assumptions in the plant licensing basis. Statistically based drift values were determined for all instruments involved except where there was insufficient drift data due to recent instrument replacement. In these cases, a 30-month drift was determined through engineering judgment considering manufacturer specifications, drift exhibited by similar devices manufactured by the same manufacturer and employed in similar applications and Westinghouse experience with similar devices. The licensee stated that such drift values will be validated through monitoring of future instrument performance. The above evaluation is consistent with GL 91-04 and, therefore, acceptable.

GL 91-04 states that plant instrument drift should be reviewed for consistency with setpoint uncertainty calculations over the extended 24-month operating cycle. The licensee in their submittal stated that the plant-specific drift data were statistically analyzed to establish instrument performance characteristics. The details of the statistical analysis of the DCPP calibration data were reported in WCAP-14646, Revision 1. To address the need for a definitive basis for drift, function specific calculations were performed to determine appropriate drift values for the sensors and process racks. The staff reviewed WCAP-14646, Revision 1, and noted that the licensee has used a three tired "graded" approach to identify the probability and confidence levels to which the drift value should be determined/varied, based on the safety significance of the instrument channel's function(s).

All of the functions included in this license amendment request for the 24-month fuel cycle fall into the RPS/ESFAS/Critical Control Category. Drift for these functions have been evaluated at a 95% probability at 95% confidence. This is consistent with the guidance of Regulatory Guide 1.105 and is, therefore, acceptable. Since the remaining two tiers were not proposed for any of the functions associated with this license amendment, the staff did not review them.

In GL-91-04, the staff identified the issues pertaining to increasing the interval of instrument surveillance and identified specific actions that licensees should take to address these issues. The staff evaluated the licensee's submittal to verify that the licensee has addressed these issues and provided an acceptable basis for increasing the calibration interval for instruments that are used to perform safety functions. Based on our evaluation as described above, the staff concludes that the licensee has confirmed that safety limits and safety analysis assumptions will not be exceeded when worst case drift has been taken into account.

3.3 Bases

The proposed changes to the Bases section support the proposed changes made for the allowable values and setpoints of the functional units of Tables 2.2-1 and 3.3-4. Therefore, the changes are acceptable.

3.4 Administrative Changes

Proposed TS Changes

Incorporate changes to the following footnotes of TS Tables 2.2-1 and 3.3-4 as noted and correct spelling of a functional unit description:

1. Functional Unit 13.b, change spelling of the word "Equivalyent" to its correct spelling Equivalent.

2. Table 2.2-1 and Table 3.3-4, T Notes 1 and 3, replace word "controller" with "compensator."

3. Table 2.2-1, Over temperature Delta T and Overpower Delta T Notes 1 and 3 clarification. Add words "loop specific" in front of word "indicated" in note 1, and add words "loop specific Indicated" between words Nominal and T_{avg} in note 3.

4. Table 2.2-1, Overpower Delta T Note 3 clarification. Add words, "Nominal Loop specific" in front of the word "Indicated".

5. TS 3/4.3.1, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements, Functional Unit 13.b - Steam Generator Water Level, Low-Low, RCS Loop Delta T input. To clarify functional unit description add words "Equivalent to Power" after words Delta T.

6. TS 3/4.3.2, Table 4.3.2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements, Functional Unit 6.c.2) - Auxiliary Feedwater, Steam Generator Water Level-Low-Low, RCS Loop Delta T Input. To clarify functional unit description, add words "Equivalent to Power" after words Delta T.

Justification for the Change

The proposed changes to the above footnotes involve a spelling correction, terminology change and clarifying language and have no safety significance. In their submittal, the licensee stated that the term "compensator" is proposed in place of "controller" to better describe signal processing in the instrument racks. The term "loop specific" is used to better define the processing that occurs for each individual RCS loop. The term "equivalent to power" is added for two items to maintain consistency in the TS for all items which refer to RCS loop Delta T equivalent to power. These changes are administrative and editorial in nature, and do not affect the design, operation, or testing of the plant. These changes are acceptable.

4.0 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.

5.0 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. 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 (62 FR 6577). 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.

6.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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