February 26, 1990

Docket Nos. 50-275
and 50-323

DISTRIBUTION:	
RegionV (4)	EJordan
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PShea	JCalvo
HRood	RJones
OC/LFMB	YHsii
GPA/PA	MChatterton

Mr. J. D. Shiffer, Vice President
Nuclear Power Generation
c/o Nuclear Power Generation, Licensing
Pacific Gas and Electric Company
77 Beale Street, Room 1451
San Francisco, California 94106

Dear Mr. Shiffer:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 73116 AND 73117)

The Commission has issued the enclosed Amendment No. 51 to Facility Operating License No. DPR-80 and Amendment No. 50 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant (DCPP), Units 1 and 2, respectively. The amendments change the Diablo Canyon combined Technical Specifications (TS) in response to your application for license amendments dated May 15, 1989, as modified by letters dated July 3, September 15, and November 30, 1989 (Reference LAR 89-06). The amendments revise the TS to allow the removal of the Boron Injection Tank (BIT) from each unit.

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

Sincerely,

ORIGINAL SIGNED BY H. ROOD

Harry Rood, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects Office of
Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 51 to DPR-80
2. Amendment No. 50 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

HR

DRSP/PD5	DRSP/PD	OGC	DRSP/(A)D:PD5
PShea*	HRood*	MYoung*	HRood
02/09/90	02/07/90	02/15/90	02/22/90

*See previous concurrence

DF01
1/1

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and 50-323

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DRSP/PD5	DRSP/PD	OGC	DRSP/(A)D:PD5
PShea	HRood		CTrammell
02/9/90	02/7/90	02/15/90	02/ /90

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

February 26, 1990

Docket Nos. 50-275
and 50-323

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

Sincerely,

Harry Rood

Harry Rood, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects Office of
Nuclear Reactor Regulation

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1. Amendment No. 51 to DPR-80
2. Amendment No. 50 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-275

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
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 May 15, 1989, as modified by letters dated July 3, September 15, and November 30, 1989, 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 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. 51, 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 becomes effective at the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Harry Rood

Harry Rood, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 26, 1990

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

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-323

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
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 May 15, 1989, as modified by letters dated July 3, September 15, and November 30, 1989, 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 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. 50, 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 becomes effective at the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Harry Rood

Harry Rood, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 26, 1990

ATTACHMENT TO LICENSE AMENDMENT NOS. 51 AND 50

FACILITY OPERATING LICENSE NOS. DPR-80 and DPR-82

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Overleaf pages are also included, as appropriate.

Remove Page	Insert Page
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-31	3/4 3-31
3/4 5-9	3/4 5-9
3/4 5-10	3/4 5-10
3/4 6-23	3/4 6-23
3/4 6-24	3/4 6-24
3/4 8-20	3/4 8-20
3/4 8-21	3/4 8-21
B 3/4 3-1	B 3/4 3-1
------	B 3/4 3-1a
B 3/4 5-2	B 3/4 5-2

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		$\geq$ 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 (Units 1 and 2 Cycle 4 and after) $\leq$ 545°F (Unit 2 Cycle 3)
	decreasing	543°F	$\geq$ 540.2°F (Units 1 and 2 Cycle 4 and after) $\geq$ 541°F (Unit 2 Cycle 3)
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	$\leq 18^{(4)}/28^{(5)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	$\leq 60^{(1)}$
6) Component Cooling Water	$\leq 38^{(4)}/48^{(5)}$
7) Containment Fan Cooler Units	$\leq 40^{(1)}$
8) Auxiliary Saltwater Pumps	$\leq 48^{(4)}/58^{(5)}$
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	$\leq$ 48.5(6)
b. Phase "B" Isolation	N.A.
c. Steam Line Isolation	$\leq$ 7
8. Steam Generator Water Level-High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 66(2)
9. Steam Generator Water Level Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
10. RCP Bus Undervoltage	
Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
11. Plant Vent Noble Gas Activity-High	
Containment Ventilation Isolation	$\leq$ 11

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	≤ 60 seconds
439	≤ 60 seconds
440	≤ 60 seconds
441	≤ 60 seconds
510	≤ 5 seconds
520	≤ 5 seconds
530	≤ 5 seconds
540	≤ 5 seconds
1510	≤ 5 seconds
1520	≤ 5 seconds
1530	≤ 5 seconds
1540	≤ 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).

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
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.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low	S	R	M	N.A	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Flow in Two Steam Lines-High Coincident With Either	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
1) T_{avg}-Low-Low, or	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2) Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-[illegible]-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, [illegible]

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons of borated water,

b. A boron concentration of between 20,000 and 22,500 ppm, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2 and 3. For Unit 1 Cycle 4, Unit 2 Cycle 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume through a recirculation flow test at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3. For Unit 1 Cycle 4, Unit 2 Cycle 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, be in HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be greater than or equal to 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
5. Power-Operated Valves (Continued)		
FCV-698*	Containment Air Sample (Post LOCA) Supply IC	N.A.
FCV-699*	Containment Air Sample (Post LOCA) Supply OC	N.A.
FCV-700*	Containment Air Sample (Post LOCA) Return OC	N.A.
PCV-19#	Steam Generator No. 1 10% Atmosphere Steam Dump OC	N.A.
PCV-20#	Steam Generator No. 2 10% Atmosphere Steam Dump OC	N.A.
PCV-21#	Steam Generator No. 3 10% Atmosphere Steam Dump OC	N.A.
PCV-22#	Steam Generator No. 4 10% Atmosphere Steam Dump OC	N.A.
8107#	Charging Line Isolation OC	N.A.
8700A#	RCS Hot Leg to RHR Pump 1 OC	N.A.
8700B#	RCS Hot Leg to RHR Pump 2 OC	N.A.
8701#	RCS Loop 4 Hot Leg to RHR IC	N.A.
8703#	RHR to RCS Hot Legs 1 and 2 IC	N.A.
8716A#	RHR to RCS Hot Legs OC	N.A.
8716B#	RHR to RCS Hot Legs OC	N.A.
8801A#	Charging Injection OC	N.A.
8801B#	Charging Injection OC	N.A.
8802A#	Safety Injection to RCS Hot Legs OC	N.A.
8802B#	Safety Injection to RCS Hot Legs OC	N.A.
8809A#	Residual Heat Removal to RCS Cold Legs 1 and 2	N.A.
8809B#	Residual Heat Removal to RCS Cold Legs 3 and 4	N.A.
8823#	Safety Injection Check Valve Test Line IC	N.A.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
5. Power-Operated Valves (Continued)		
8824#	Safety Injection Check Valve Test Line IC	N.A.
8843#	Charging Injection IC	N.A.
8835#	Safety Injection to RCS Cold Legs OC	N.A.
8885A#	RHR to Cold Leg Test Line IC	N.A.
8885B#	RHR to Cold Leg Test Line IC	N.A.
8982A#	Containment Sump to Residual Heat Removal Train 1 OC	N.A.
8982B#	Containment Sump to Residual Heat Removal Train 2 OC	N.A.
8980#	Refueling Water Storage Tank to RHR OC	N.A.
9001A	Containment Spray Pump No. 1 Isolation OC	N.A.
9001B	Containment Spray Pump No. 2 Isolation OC	N.A.
9003A#	Residual Heat Removal to Containment Spray OC	N.A.
9003B#	Residual Heat Removal to Containment Spray OC	N.A.
6. Check Valves		
8028	Relief Valve Outlets to Pressurizer Relief Tank IC	N.A.
8046	Primary Water to Pressurizer Relief Tank IC	N.A.
8047	Nitrogen to Pressurizer Relief Tank IC	N.A.
8109	Seal Water Return IC	N.A.
8368A thru 8368D	Seal Water to Reactor Coolant Pumps IC	N.A.
8916	Nitrogen Supply to Accumulators IC	N.A.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 The thermal overload protection and bypass devices, integral with the motor starter, of each valve listed in Table 3.8-1 shall be OPERABLE.

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

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valves.

SURVEILLANCE REQUIREMENTS

4.8.4.1 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

   1) Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

   2) Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

   1) All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years, and

   2) All thermal overload devices which are continuously bypassed, such that each continuously bypassed device is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload device is OPERABLE and not bypassed at least once per 6 years.

TABLE 3.8-1

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
8107	Charging Isolation	Yes
8105	Charging Pumps' Recirculation	Yes
FCV-430	Component Cooling Heat Exch. Outlet	No
LCV-112B	Volume Control Tank Outlet	Yes
8801A	Charging Injection	Yes
8803A	Charging Injection	Yes
8807A	Safety Injection/Charging Suction Crosstie	No
8805A	RWST to Charging Pumps	Yes
FCV-437	Raw Water Supply to Auxiliary Feedpumps	No
FCV-441	S/G 4 Feedwater Isolation	Yes
FCV-750	RCP Thermal Barrier CCW Return	Yes
FCV-438	S/G 1 Feedwater Isolation	Yes
8923A	SI Pump 1 Suction	No
FCV-38	Aux. FWP Turb. Steam Supply	No
8980	RWST to RHR	No
8974A	SI Pumps' Recirculation	No
8992	Spray Additive Tank Outlet	Yes
8000A	Pressurizer RV Isolation	No
FCV-601*	Auxiliary Saltwater Pumps Crosstie	No
8808A	Accumulator 1 Isolation	No
8802A	SI Pump 1 to Hot Leg	No
8821A	SI Pump 1 to Cold Legs	No
8808D	Accumulator 4 Isolation	No
8808B	Accumulator 2 Isolation	No
8106	Charging Pumps' Recirculation	Yes
8108	Charging Isolation	Yes
LCV-112C	Volume Control Tank Isolation	Yes

*FCV-601 is common to both units.

TABLE 3.8-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
8809A	RHR to Cold Legs	No
8801B	Charging Injection	Yes
8805B	RWST to Charging Pumps	Yes
8700A	RHR Pump 1 Suction	No
8804A	RHR to Charging Pumps	No
FCV-436	RWSR to Auxiliary Feedpump	No
9001A	Spray Isolation	Yes
FCV-363	RCP CCW Return Isolation	Yes
8835	SI Pumps to RCS Cold Legs	No
8701	RCS to RHR System	No
8100	RCP Seal Water Return	Yes
8803B	Charging Injection	Yes
FCV-431	CCW Heat Exchanger Outlet	No
FCV-641A	RHR Recirculation	No
8716A	RHR to Hot Legs	No
8000B	Pressurizer RV Isolation	No
FCV-439	S/G 2 Feedwater Isolation	Yes
9003A	RHR to Spray	No
FCV-95	Turb. Feedpump Steam Supply	Yes
8994A	Spray Additive Tank Outlet	Yes
8703	RHR to Hot Legs	No
8104	Emergency Borate	No
8982A	Containment Sump RHR Recirculation	No
LCV-106	S/G 1 Aux. Feedwater Supply	No
LCV-107	S/G 2 Aux. Feedwater Supply	No
LCV-108	S/G 3 Aux. Feedwater Supply	No
LCV-109	S/G 4 Aux. Feedwater Supply	No

TABLE 3.8-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
FCV-356	RCP CCW Supply Isolation	Yes
8808C	Accumulator 3 Isolation	No
FCV-641B	RHR Recirculation	No
FCV-355	CCW Header C Isolation	Yes
9001B	Spray Isolation	Yes
8982B	Containment Sump RHR Recirculation	No
9003B	RHR to Spray	No
8976	RWST to SI Pumps	No
8804B	RHR to SI Pumps	No
8802B	SI Pump 2 to Hot Legs	No
8112	RCP Seal Water Return Isolation	Yes
FCV-357	RCP Barrier CCW Return Isolation	Yes
FCV-749	RCP Bearing Cooling H_2O Return Isolation	Yes
8702	RCS to RHR Suction	No
FCV-440	S/G 3 Feedwater Isolation	Yes
8716B	RHR to RCS Hot Legs	No
FCV-37	Aux. Feedpump Steam Supply	No
8821B	SI Pump 2 to Cold Legs	No
8807B	Safety Injection/Charging Suction Crosstie	No
8000C	Pressurizer RV Isolation	No
8994B	Spray Additive Tank Outlet	Yes
FCV-495	ASW Crosstie	No
FCV-496	ASW Crosstie	No
8700B	RHR Pump 2 Suction	No
8974B	SI Pumps Recirculation	No
8809B	RHR to Cold Legs	No
8923B	SI Pump 2 Suction	No

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 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 Engineered Safety Features Actuation System 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 Engineered Safety Features Actuation System 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 automatic valves position, and (11) component cooling water pumps start and automatic valves position.

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

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

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.

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

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

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

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on Low Pressurizer Pressure.

P-12 On increasing reactor coolant loop temperature, P-12 automatically reinstates Safety Injection actuation on High Steam Flow coincident with either Low-Low T_{avg} or Low Steam Line Pressure, and provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 allows the manual block of Safety Injection actuation on High Steam Flow coincident with either Low-Low T_{avg} or Low Steam Line Pressure and automatically removes the arming signal from the Steam Dump System.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

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

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

3/4.5.1 ACCUMULATORS

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

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

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

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

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

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

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

ECCS SUBSYSTEMS (Continued)

The requirement to maintain the RHR Suction Valves 8701 and 8702 in the locked closed condition in MODES 1, 2 and 3 provides assurance that a fire could not cause inadvertent opening of these valves when the RCS is pressurized to near operating pressure. These valves are not part of an ECCS subsystem.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all centrifugal charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable below 323°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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

3/4.5.4 BORON INJECTION SYSTEM

The Boron Injection System is only required for Unit 1 Cycle 4 and Unit 2 Cycle 3. The OPERABILITY of the Boron Injection System as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21,000 ppm boron.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NO. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated May 15, 1989, as modified by letters dated July 3, September 15, and November 30, 1989 (Reference LAR 89-06), Pacific Gas and Electric Company (PG&E or the licensee) requested amendments to the combined Technical Specifications (TS) appended to Facility Operating License Nos. DPR-80 and DPR-82 for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2, respectively. The amendments change the TS to allow the removal of the boron injection tank (BIT). Specifically, the license amendment request proposed to:

(1) Limit the applicability of Technical Specification (TS) 3/4.5.4, "Boron Injection System" and the associated bases to Unit 1 Cycle 4 and Unit 2 Cycle 3 operation. After those cycles, TS 3/4.5.4 and the associated bases will no longer apply,

(2) Revise TS Table 3.3-5, "Engineered Safety Features Response Times," by increasing the safety injection (SI) response times to 25 and 35 seconds for the offsite power available and unavailable cases, respectively, to be consistent with the BIT removal program, and

(3) Revise TS Table 3.8-1, "Motor-Operated Valves Thermal Overload Protection and Bypass Devices," and TS Table 3.6-1, "Containment Isolation Valves," by changing the functional description of the BIT inlet and outlet valves and other valves to charging injection valves.

The staff evaluation of these changes is given below.

2.0 EVALUATION

The NRC staff has evaluated the proposed changes and finds them acceptable, based on its review of the analyses and evaluations given by the licensee. A discussion of each of the specific technical specification changes made by these amendments is presented below.

The BIT was originally incorporated into Westinghouse-designed plants as a means of mitigating the consequence of accidental depressurization of the main steam system events. The sole purpose of the BIT, as a component of the Safety Injection System, is to provide concentrated boric acid (20,000 ppm) and thus a negative reactivity insertion during accidents. Problems and safety concerns associated with the BIT were identified in NRC Generic Letter 85-16, "High Boron Concentrations," August 23, 1985. The use of high concentration boric acid imposes operational and maintenance problems such as minimum volumes and concentrations in boric acid system tanks, heat tracing malfunctions, BIT valve testing, and recovery from inadvertent safety injection, which adversely affect plant availability. The high boric acid concentrations also cause safety concern involving boric acid solidification which may render the emergency core cooling inoperable. Therefore, many plants such as Beaver Valley, Byron/Braidwood, Turkey Point, McGuire and Catawba have removed the BITs. For these reasons, PG&E also decided to remove the BIT and the associated heat tracing systems from Diablo Canyon during the next refueling outage of Units 1 and 2.

The proposed change to limit the applicability to Unit 1 Cycle 4 and Unit 2 Cycle 3 operations for TS 3/4.5.4 is necessary to reflect the planned removal of the BIT and associated heat tracing systems. Since the BIT removal can only be implemented during a refueling outage, the cycle-specific TS changes reflect the differences between Units 1 and 2 that will exist between the Unit 2 third refueling outage and the Unit 1 fourth refueling outage. Thereafter, TS 3/4.5.4 can be completely deleted. Revisions of the functional description of certain valves in TS Tables 3.6-1 and 3.8-1 is necessary to properly reflect the removal of the BIT.

The proposed BIT removal also requires a change in the safety injection response times. The increase of the SI response times is due to the interlock logic between the refueling water storage tank (RWST) and volume control tank (VCT) outlet isolation valves which ensures a water source to the suction of the centrifugal charging pumps, but delay the delivery of borated water to the RCS by the valve stroke time of the VCT outlet isolation valves. Since the BIT is downstream of the charging pumps, borated water from the BIT would be delivered to the RCS regardless of where the charging pump suction source was coming from, and no TS changes were required as long as the BIT boron concentration requirement was in effect. However, the proposed elimination of the BIT boron concentration requirement necessitates the revision of the safety injection response times. The extra safety injection response times are included to account for the sequential operation of the refueling water storage tank and the closing of the Volume Control Tank valve. Therefore, the response times for delivery of the 2300 ppm borated water to the primary system in TS Table 3.3-5 are increased to 25 seconds and 35 seconds, respectively, for the off-site power available and unavailable situations. Both the BIT removal and the SI response time increase are supported by the safety analysis using the new response times for the cases where the offsite power is available and not available to ensure compliance with the safety limits and other acceptance criteria.

The licensee, by letter dated September 15, 1989, submitted WCAP-11938, Volume 1, "BIT Elimination Study for Diablo Canyon Units 1 and 2," which provides safety analysis with the removal of the BIT. The safety analysis was performed for (1) the hypothetical steamline break, with and without offsite power available, for the largest double ended rupture of a steam pipe, and (2) credible steamline break, with and without offsite power available, for the largest single failed open steam generator relief, safety and dump valve. The credible steamline break is an ANS Condition II event with the acceptance criteria that the specified acceptable fuel design limits should not be violated. The hypothetical main steamline break is an ANS Condition IV event with the criteria that the radiological release should not exceed the limits set forth in 10 CFR 100.

The steamline breaks were analyzed using the NRC approved method and computer code, LOFTRAN. Conservative assumptions and initial conditions were used in the analyses. For example, the analysis was set up to conservatively account for a low steamline pressure setpoint of 15 psia. Even though the BIT recirculation lines between the BIT and the boric acid tanks along with the isolation valves and flow instrument will be cut out and removed, the BIT and associated piping are conservatively modeled as being in place and full of water having 0 ppm boric acid. The safety injection (SI) response time for delivering the 2300 ppm borated water to the primary system was modeled as a 22 second pure delay followed by a 10 second linear ramp in the SI flow for the cases with the offsite power available. This is different from the 25 seconds in the proposed TS changes. However, since the SI delay time in the TS is defined as the duration between the time when the process parameters being measured reach the setpoint and the time when the charging pumps are at full flow, the assumed 22 second pure time delay and a 10 second linear ramp of SI flow in the safety analysis is considered adequate.

The steamline break analysis used the W-3 critical heat flux correlation with an approved minimum departure from nucleate boiling ratio (DNBR) limit of 1.45 for the pressure below 1000 psia. Since a homogeneous core was used in the analysis for the transitional mixed core of the standard fuel and the Vantage 5 fuel, a mixed core penalty calculated with approved method was assigned to the Vantage 5 fuel which has higher hydraulic resistance than the standard fuel and would divert flow into the standard fuel if a true mixed core was modeled in the analysis. No mixed core penalty was assigned to the standard fuel because of its lower hydraulic resistance. The resulting DNBRs in all cases analyzed never fall below the minimum DNBR limit. Therefore, there is a 95 percent probability at 95 percent confidence level that DNB will not occur and there is no fuel failure. Even for the ANS Condition IV main steamline break event, no fuel failure is anticipated and the radiological consequence complies with the 10 CFR 100 criteria.

In summary, the staff has reviewed the safety analysis performed to support the proposed TS changes for removal of BIT and the associated heat tracing systems, as well as the increase in the SI response time. Since approved methods were used for the analysis and the results conform to the acceptance criteria, the proposed TS changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and a change in surveillance requirements. At Diablo Canyon, the restricted area coincides with the site boundary. We have 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. Accordingly, these 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 these amendments.

4.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and (3) the issuance of these amendments will not be inimical to the common defense and security or the health and safety of the public.

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Dated: February 26, 1990