August 19, 1996

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. M90971) AND UNIT NO. 2 (TAC NO. M90972)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. DPR-80 and Amendment No. 113 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 November 14, 1994, and supplemented by letters dated December 7, 1995, February 2, 1996, May 28, 1996, and July 30, 1996.

These amendments revised the TS for the slave relay test frequency from quarterly (Q) to refueling (R), based on generic Westinghouse Owners Group (WOG) topical reports. The revision also removed table notation 4 from Table 4.3-2. The associated Bases were also appropriately revised.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next 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. 115 to DPR-80
2. Amendment No. 113 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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DOCUMENT NAME: DC90971.AMD

* See previous concurrence

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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. 115
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 November 14, 1994, as supplemented by letters dated December 7, 1995, February 2, 1996, May 28, 1996, and July 30, 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. 115 , 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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: August 19, 1996

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. 113
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 November 14, 1994, as supplemented by letters dated December 7, 1995, February 2, 1996, May 28, 1996, and July 30, 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. 113, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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: August 19, 1996

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 113 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
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 3-1a	B 3/4 3-1a

TABLE 3.3-5 (Continued)

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.	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)	R	1, 2, 3, 4
c. Containment Pressure-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low	S	R	Q	N.A	N.A.	N.A.	N.A.	1, 2, 3
e. DELETED								
f. Steam Line Pressure-Low	S	R	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.	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)	R	1, 2, 3, 4
c. Containment Pressure-High-High	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
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	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.	R	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	R	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.	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. Containment Pressure-High-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Negative Steam Line Pressure Rate-High	S	R	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	R	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	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3(5)
2) RCS Loop ΔT	N.A.	R	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 CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL 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.	R	N.A.	R	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.	R	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.	R	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.

BASES

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 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 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 (11) component cooling water pumps start and automatic valves position.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the trips are set for each functional unit. 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 accuracy. For all setpoints in digital hardware, the setpoints are set at the nominal values.

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

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.

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.

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 on-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. 115 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

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

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By application dated November 14, 1994, as supplemented by letters dated December 7, 1995, February 2, 1996, May 28, 1996, and July 30, 1996, Pacific Gas and Electric Company (or the licensee), as the lead plant, 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 (DCPP). The proposed changes, based on generic Westinghouse Owners Group (WOG) topical reports, would revised the slave relay test frequency from quarterly (Q) to refueling (R). Currently at DCPP and other Westinghouse plants, slave relays for the engineered safety features actuation system (ESFAS) are tested quarterly with the exception of some relays which were previously approved by the NRC to be tested every 18 months. The proposed changes to the TS would extend the test interval for all Potter and Brumfield MDR slave relays in Westinghouse plant ESFAS to 18 months. In order to justify these changes, PG&E provided generic Westinghouse Topical Reports, WCAP-13878, Rev. 0, "Reliability & Assessment of Potter and Brumfield MDR Series Relays," dated June 1994, (proprietary version) (Ref. 1), WCAP-14117, Rev. 0, dated June 1994, (non-proprietary version), (Ref. 2) and WCAP-13900, Rev. 0, "Extension of Slave Relay Surveillance Test Intervals," dated April 1994 (Ref. 3). Following review of the above topical reports, the staff, by letter dated April 25, 1995 (Ref. 4) requested additional information and PG&E responded by letters dated December 7, 1995 and February 2, 1996 (Refs. 5 and 6). In addition, by letter dated April 12, 1996, the Westinghouse Owners Group submitted Revision 1 to WCAP-13878 and WCAP-14117 (Ref. 7). By letter dated May 31, 1996, the staff accepted and issued a Safety Evaluation Report (SER) to the WOG approving the above Topical Reports. The revision also removed table notation 4 from Table 4.3-2 and revised the appropriate associated Bases.

The December 7, 1995, February 2, 1996, May 28, 1996, July 30, 1996, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration noticed in the Federal Register on December 6, 1995 (60 FR 62495).

2.0 BACKGROUND

The NRC staff formed a Task Group in August 1983 to investigate problems concerning surveillance testing required by TS and to recommend improvements. The results of the study were published in November 1983 (Ref. 8) in NUREG-1024, "Technical Specifications - Enhancing the Safety Impact." NUREG-1024 recommended that the staff review the bases for TS test frequencies; ensure that the TS required tests promote safety and do not degrade equipment; and review surveillance tests to ensure that they do not unnecessarily burden personnel.

The Technical Specifications Improvement Program (TSIP) was established in December 1984 to provide the framework for addressing the NUREG-1024 recommendations, and for rewriting and improving the TS. As an element of the TSIP, TS surveillance requirements were comprehensively examined as recommended in NUREG-1024. The results of the TSIP effort are presented in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements" (Ref. 9). The study concluded that, while some testing at power is essential, safety can be improved, equipment degradation decreased, and unnecessary personnel burden prevented by reducing the amount of testing at power. These three conclusions formed the basis for the four criteria that justify changes to surveillance intervals as follows:

Criterion 1 - The surveillance could lead to a plant transient,

Criterion 2 - The surveillance results in unnecessary wear to equipment,

Criterion 3 - The surveillance results in radiation exposure to plant personnel that is not justified by the safety significance of the surveillance,

Criterion 4 - The surveillance places an unnecessary burden on plant personnel because the time required is not justified by the safety significance of the surveillance.

3.0 EVALUATION

In the SER for the above Westinghouse Topical Reports, the staff requested each licensee to address the following concerns:

1. Confirm the applicability of the WCAP-13878, Rev. 1 analyses for their plant.

2. Ensure that their procurement program for P&B MDR relays is adequate for detecting the types of failures that are discussed in References 10, 11, 12 and 13.

3. Ensure that all pre-1992 P&B MDR relays which are used in either normally energized or a 20 percent duty cycle have been removed from ESFAS applications.

4. Ensure that a contact loading analysis for P&B MDR relays has been performed to determine the acceptability of these relays.

The DCPP licensee in its submittal dated December 7, 1995, addressed each of the above issues. The licensee's response to these issues is discussed below:

1. Slave relays used at DCPP are P&B MDR Model 4102 (latching) and 4103 (non-latching) type. Although WCAP-13878 analyzed P&B MDR model 4121-1 (latching) and 4103 (non-latching) type relays, the DCPP relays are similar in design to those analyzed in the WCAP, and therefore, the analysis adequately covers the DCPP relays. The staff concurs with the licensee's statement on the similarity of relays.

2. Based on Report S93-06 from the NRC Office for Analysis and Evaluation of Operational Data (AEOD) (Ref. 10), Information Notice (IN) 90-57 and Supplement 1 to IN 90-57 (Refs. 11 and 12), and a 10 CFR Part 21 notification from San Onofre Nuclear Generating Station (Ref. 13), PG&E has put in place an enhanced commercial grade dedication program to prevent substandard or refurbished relays from being installed in the plant. All existing relays in the warehouse will be re-inspected based on the enhanced dedication criteria. Only three relays procured as commercial grade were installed at DCPP and they have been verified to have passed sufficient dedication criteria and testing to assure their acceptability. Based on this, the staff finds that PG&E has an adequate commercial grade dedication program for P&B MDR relays for detecting potential failures.

3. IN 92-04 (Ref. 14) and AEOD Report S93-06, identified failures of normally energized or 20 percent duty cycle P&B MDR relays. At DCPP, there are 40 P&B MDR relays installed in each of the solid state protection system (SSPS) bays. Of these 40 relays, only 2 relays are normally energized at power, and neither relay performs a function covered by TS. Also at DCPP, none of the slave relays required by TS are energized during an outage since the SSPS is removed from service at that time. Therefore, the concern for removing normally energized or 20 percent duty cycle relays from ESFAS applications does not apply to DCPP.

4. IN 92-19 (Ref. 15) and AEOD Report S93-06 reported contact failures due to misapplication of P&B MDR relays. WCAP-13878, Rev. 0, considered contact failures to be beyond the scope of the report. However, PG&E completed a loading study covering each contact on every DCPP SSPS P&B MDR slave relay for Unit 1 and found the relays acceptable. The DCPP Unit 2 design and contact loading is similar to Unit 1 and hence was not reviewed. The licensee's analysis satisfactorily resolves the staff's concern in this area.

Based on the review of WCAP-13878, Rev. 1, WCAP-14117, Rev. 1, and WCAP-13900, Rev. 0, and the licensee's submittals referencing these topical reports, the staff concludes that the proposed test interval extension to every refueling outage for P&B MDR slave relays is justified for DCPP. However, consistent with the staff SER for the above topical reports, the staff further concludes that if two or more P&B MDR ESFAS subgroup relays fail in a 12-month period, the licensee should reevaluate the adequacy of the extended surveillance interval. The reevaluation should consider design, maintenance and testing of all P&B MDR ESFAS subgroup relays. If the licensee determines that the surveillance interval is inadequate for detecting a single relay failure, the surveillance interval should be decreased. The revised surveillance interval should be such that the licensee can detect an ESFAS subgroup relay failure prior to the occurrence of a second failure. By letter dated July 30, 1996, the licensee committed to reevaluate the adequacy of the extended surveillance interval according to the criteria presented in the generic SER.

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 changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 62495). 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.

7.0 REFERENCES

1. Westinghouse Topical Report WCAP-13878, "Reliability Assessment of Potter & Brumfield MDR Series Relays," (proprietary version) dated June 1994, transmitted to NRC by Gregory M. Rueger (Pacific Gas and Electric Company for Diablo Canyon) letter DCL-94-254, dated November 14, 1994.

2. Westinghouse Topical Report WCAP-14117, "Reliability Assessment of Potter & Brumfield MDR Service Relays," (non-proprietary version) dated June 1994, transmitted to NRC by Gregory M. Rueger (Pacific Gas and Electric Company for Diablo Canyon) letter DCL-94-254, November 14, 1994.

3. Westinghouse Topical Report WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," dated April 1994, transmitted to NRC by Gregory M. Rueger (Pacific Gas and Electric Company for Diablo Canyon) letter DCL-94-254, dated November 14, 1994.

4. Melanie A. Miller (NRC) letter to Gregory M. Rueger (PG&E), dated April 27, 1995, "Request for Additional Information on Slave Relay Test Frequency Extension for Diablo Canyon Nuclear Power Plant, Units 1 and 2."

5. Gregory M. Rueger (PG&E) letter (DCL-95-268) to USNRC, dated December 7, 1995, "Response to NRC Request for Additional Information on Slave Relay Test Frequency Relaxation Amendment."

6. Warren H. Fujimoto (PG&E) letter (DCL-96034) to USNRC, dated February 2, 1996, "Respond to NRC Request on Slave Relay Test Frequency Relaxation Amendment."

7. Lee Bush (WOG) letter (WOG-SRT-96-005) to USNRC, dated April 12, 1996, "Transmittal of Page Revisions to WCAP-13878 (proprietary), to address NRC review issues.

8. NUREG-1024, "Technical Specifications - Enhancing the Safety Impact," dated November 1983.

9. NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," dated December 1992.

10. Office for Analysis and Evaluation of Operational Data Special Study Report AEOD/S93-06, "Potter & Brumfield Model MDR Rotary Relay Failures," dated December 1993.

11. NRC Information Notice 90-57, "Substandard, Refurbished Potter & Brumfield Relays Misrepresented as New," dated September 5, 1990.

12. NRC Information Notice 90-57, Supplement 1, "Substandard, Refurbished Potter & Brumfield Relays Represented as New," dated November 27, 1991.

13. 10 CFR Part 21 Notification dated July 21, 1995 from San Onofre Nuclear Generating Station (SONGS) concerning relays that were returned to SONGS with bent contact arms following PLB rework.

14. NRC Information Notice 92-04, "Misapplications of Potter & Brumfield MDR Model MDR Rotary Relay Failures," dated January 6, 1992.

15. NRC Information Notice 92-19, "Misapplications of Potter & Brumfield MDR Rotary Relays," dated March 2, 1992.

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