50-275

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

October 1, 1995

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 AND UNIT NO. 2

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-80 and Amendment No. 107 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 September 30, 1995.

These amendments increase the setpoint tolerance of the main steam safety valves (MSSVs) from ±1 percent to ±3 percent, with the exception that the lowest set MSSVs would have a tolerance of -2 percent/+3 percent.

Your letter of September 30, 1995, requested that this amendment be treated as an emergency because recent testing has shown that MSSVs have a lift distribution specific to each valve that may exceed ±1 percent.

The staff is concerned regarding the variability and magnitude of some of the MSSV lifts identified during initial testing. Based on a teleconference with the licensee on October 1, 1995, the licensee committed to perform augmented inspection and testing on the MSSVs on a more frequent basis than that required by the ASME Code for at least one operating cycle. Additional information on the details of the inspection and testing program will be provided to the staff by November 1, 1995.

CP-1

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 K. Thomas for

James C. Stone, Senior 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. 108 to DPR-80
2. Amendment No. 107 to DPR-82
3. Safety Evaluation

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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,

Kirsten M. Thomas for

James C. Stone, Senior 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:
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3. Safety Evaluation

cc w/encls: See next page

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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. 108
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 September 30, 1995, 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. 108, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

William H. Bateman

William H. Bateman, Director
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: October 1, 1995

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. 107
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 September 30, 1995, 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. 107, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

William H. Bateman

William H. Bateman, Director
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 1, 1995

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 107 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 7-3	3/4 7-3
B 3/4 7-1	B 3/4 7-1

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

LIFT SETTING*	ORIFICE SIZE
1065 psig (-2%, +3%)**	4.515 inches
1078 psig (±3%)**	4.515 inches
1090 psig (±3%)**	4.515 inches
1103 psig (±3%)**	4.515 inches
1115 psig (±3%)**	4.515 inches

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

**Within ±1% following main steam line Code safety valve testing.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate vital busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from two OPERABLE and redundant steam supply sources.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Deleted.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The primary purpose of the main steam safety valves (MSSVs) is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary by providing a heat sink for the removal of energy from the reactor coolant system if the preferred heat sink, provided by the condenser and the circulating water system, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves. The MSSV design includes staggered setpoints, according to Table 3.7-2 so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

The design basis for the MSSVs comes from ASME Code Section III, and its purpose is to limit the secondary system pressure to less than or equal to 110% of design pressure. This design basis is sufficient to cope with any anticipated operational occurrence or accident considered in the design basis accident and transient analysis. The tolerance on the MSSV setpoints assures that the secondary system will not be overpressurized if the MSSVs lift at the high end of their tolerance band, and assures that the steam generators (SGs) will not be overfilled during a SG tube rupture if the MSSVs lift at the low end of their tolerance band. A minimum of two OPERABLE safety valves per SG ensures that sufficient relief capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint,

X = Total relieving capacity of all safety valves per steam line in lb/hour, and

Y = Maximum relieving capacity of any one safety valve in lb/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

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

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 440 gpm at a pressure of 1135 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 880 gpm at a pressure of 1135 psig to the entrance of the steam generators. The capacity of one motor-driven AFW pump (440 gpm) delivered to at least two steam generators is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 AUXILIARY FEEDWATER SOURCE

The principal function of the Auxiliary Feedwater (AFW) Source is to provide a qualified source of water to the steam generators via the AFW System for removal of decay and sensible heat from the Reactor Coolant System (RCS) through generation and release of steam.

The minimum usable water volume in the Condensate Storage Tank (CST) ensures the availability of sufficient water for cooldown of the RCS to less than 350°F in the event of a total loss of offsite power. This minimum volume is also sufficient to remove decay heat sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to the atmosphere.

An alternate plant cooldown scenario has been postulated for the loss of offsite power, which assumes a reduced Reactor Coolant System cooldown rate and provides credit only for seismically qualified water sources. The lower rate increases the cooldown time period until the Residual Heat Removal System can be used to remove further decay heat. The capacity of the seismically qualified portion of the CST is less than the total amount of water needed for the extended time period. The Fire Water Storage Tank (FWST) has been identified as the seismically qualified source of additional water in the event of an extended cooldown without offsite power.

With the CST less than the required volume, the volume must be restored to the limit. Four hours provides time to restore the required volume from the condenser, or other source, and is a reasonable time to limit the risk from accidents requiring the plant to cool down.

With the FWST unable to supply the required backup volume of cooling water to the AFW System, the operability of the supply must be restored within seven days. This is considered a reasonable time to limit the risk of an accident which would require the use of the backup volume in addition to the primary volume maintained in the CST. Alternate non-seismically qualified water sources are also available to supply water to supplement the CST volume.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 107 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated September 30, 1995, Pacific Gas and Electric Company (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 on an emergency basis. The proposed changes would revise Table 3.7-2 of Technical Specification (TS) 3.7.1.1 and the associated Bases to increase the setpoint tolerance of the main steam safety valves (MSSVs) from ±1 percent to ±3 percent, with the exception that the lowest-set MSSVs would have a tolerance of -2 percent/+3 percent. The proposed amendment would allow Diablo Canyon Unit 2 to restart following an unanticipated shutdown on September 23, 1995.

The safety evaluation for a license amendment request (LAR) to increase the setpoint tolerance for the MSSVs was initiated several years ago by the licensee in recognition of generic industry MSSV setpoint concerns. Prior to submitting this emergency license amendment request, the licensee was in the process of submitting the safety evaluation as part of an LAR to revise the MSSV setpoint tolerance. However, as a result of recent (September 12 to September 29, 1995) extensive MSSV testing, the licensee has determined that the MSSVs have a setpoint lift distribution specific to each valve. In addition, the licensee determined that the distribution for a specific valve may exceed ±1 percent.

2.0 DISCUSSION AND EVALUATION

Overpressure protection of the main steam system is provided by five MSSVs located on each of the four main steam lines. There are a total of 20 MSSVs on each unit, set at staggered setpoints (1065, 1078, 1090, 1103 and 1115 psig), with sufficient capacity to limit secondary system pressures to less than 110 percent of the system design pressure.

The plant inservice testing (IST) program for the MSSVs currently meets the requirements of the 1977 Edition through 1978 Addenda of Section XI of the American Society of Mechanical Engineers (ASME) Code which does not specify a tolerance to be applied to the MSSV setpoint test results. The current TS 3.7.1.1 tolerance of ±1 percent has been used as the acceptance criterion for ASME Code Section XI testing. Section XI requires a portion of the valves be tested each operating cycle with all being tested within five years and that additional MSSV testing be performed when a MSSV fails a setpoint test. This schedule will continue to be used except that the criterion for testing additional valves will be ±3 percent (-2, +3 percent for the lowest-set MSSVs) of the nominal setpoint instead of ±1 percent. However, all valves that are tested and found to be outside the ±1 percent band will be reset to within ±1 percent of the nominal setpoint. If a valve is found outside ±3 percent (-2, +3 percent for the lowest-set MSSVs), corrective actions and a safety assessment will also be required. The licensee's proposed TS and IST criteria are consistent with the 1989 Edition of ASME Code Section XI, which allows the as-found setpoint to exceed the nominal value by up to 3 percent.

The licensee recently changed the onsite testing methodology to use the AVK "Ultra-Star" methodology which is a hydraulically-assisted method similar to the Trevitest method previously used, except that the lift point is determined by an acoustic sensor and the system pressure is determined by a pressure transducer. The licensee stated that the AVK method thereby minimizes the potential for human error. As a result of an array of testing performed on the plant MSSVs and extensive correlations between the licensee's test method and actual laboratory steam tests, the licensee has determined that the variance of the MSSV setpoints is greater than the current ±1 percent TS tolerance. The licensee found that the expected variance for both their own testing methodology and laboratory steam testing is similar and is within approximately ±3 percent using one standard deviation of the test result data. The licensee further determined that testing experience has shown that the initial lift can be expected to be at a slightly higher pressure than immediate subsequent lifts. The licensee has determined that there were several tests where the initial lifts exceeded the nominal setpoint by plus 3 percent after a period of power operation, but that for greater than 50 percent of the tests, the second lifts were within ±1 percent of the nominal setpoints. The licensee postulates that once the initial bond of the valve disk to its seat is broken, the valve setpoint will revert to a value closer to its nominal value. As described below, there is sufficient conservatism in the system overpressure analyses such that if the initial MSSV lifts were slightly higher consistent with a greater amount of setpoint variance, the peak pressures would not exceed 110 percent of design pressure.

To support the proposed TS amendment for the change to the MSSV setpoint tolerances, the licensee has performed an evaluation and analyses to determine the impact on the design basis transients and accidents for Diablo Canyon Nuclear Power Plant, Units 1 and 2. All of the transients and accidents documented in the Updated FSAR were evaluated by the licensee to determine the impact of the proposed change to the TS. For the cases where the TS change had an adverse impact on the event consequences, a detailed evaluation or reanalysis of the event has been performed.

The licensee used the RETRAN computer code to perform its reanalyses of the MSSV setpoint tolerance change. A description of the licensee's RETRAN model and a comparison of its calculated values with plant transient data and Updated FSAR data for a turbine trip were submitted along with the TS change request. The data comparison indicates that the RETRAN calculation overpredicts the peak system pressures during the most limiting heatup transients. The assumptions used in the RETRAN code are essentially the same as those used in the Updated FSAR. Based on the staff's review of the licensee's submittal, the staff believes that the use of the licensee's RETRAN computer code could reasonably estimate the peak transient system pressures and could, therefore, be used to perform an assessment for the impact of the TS change regarding MSSV setpoint tolerances.

The licensee has identified the loss of external electrical load/turbine trip as the limiting transient regarding peak primary and secondary system pressures. The licensee has reanalyzed cases for this transient with and without pressurizer pressure control. The results of its analysis indicate that the most limiting peak primary pressure of 2743 psia occurs in the case without pressurizer pressure control and the most limiting peak secondary system pressure of 1183 psia occurs in the case with pressurizer pressure control. In both cases, the peak transient primary and secondary system pressures are within 110 percent of the system design pressures and thus the results are acceptable to the staff. The licensee's evaluation concluded that the impact of the proposed TS change on all other heatup transients and accidents are bounded by the consequences of the turbine trip transient. Based on its review of the licensee's submittal, the staff agrees with the licensee's evaluation.

The licensee has evaluated the effects of the TS change to a small break loss-of-coolant accident (LOCA) and determined that a penalty of 117 degrees F will be added to the FSAR documented peak clad temperature (PCT) of 1246 degrees F and, therefore, based on an upper limit of 2200°F, sufficient safety margin still exists. The licensee also evaluated the consequences of a steam generator tube rupture event and concluded that the change to the MSSV setpoint tolerance would not have a significant effect on the radiological consequences. This is because the majority of the steam will be released from the atmospheric steam dump valves, which are operated at lower pressures.

The staff has reviewed the results of the licensee's assessment and agrees with its conclusion. Therefore, the staff finds the proposed change to the MSSV setpoint tolerance acceptable. However, the staff is concerned regarding the variability and magnitude of some of the MSSV lifts identified during initial testing. Based on a teleconference with the licensee on October 1, 1995, the licensee committed to perform augmented inspection and testing on the MSSVs on a more frequent basis than that required by the ASME Code for at least one operating cycle. Additional information on the details of the inspection and testing program will be provided to the staff by November 1, 1995.

3.0 EMERGENCY CIRCUMSTANCES

On September 14, 1995, MSSV testing was completed in preparation for the Unit 1 refueling outage. The as-found lift setpoints for 19 of 20 Unit 1 MSSVs were determined to be outside the TS 3.7.1.1 allowable setpoint tolerance of ±1 percent. Of these 19 valves, 16 lifted between 3 percent and 9 percent above their nominal setpoints. The valves were returned to within ±1 percent as required by the TS.

In response to the Unit 1 MSSV test results, testing of the Unit 2 MSSVs was initiated on September 21, 1995. Prior to completing the testing, a Unit 2 manual reactor trip occurred on September 23, 1995. Of the 16 valves tested prior to the event, 11 lifted greater than 1 percent above their setpoint, five lifted greater than 3 percent above their setpoint, and one lifted greater than 3 percent below its setpoint.

During the Unit 2 trip, two of the lowest-set MSSVs lifted when condenser vacuum was lost. These valves, which were 1065 psig setpoint valves, lifted at approximately 1028 and 1023 psig. Later in the event, a third 1065 psig setpoint valve lifted at approximately 1047 psig, and investigation is continuing into the possible lift of another MSSV.

Following evaluation of the Unit 2 MSSV test and trip results, the licensee undertook an extensive MSSV testing program, including the testing of all 20 MSSVs, to further evaluate this information. Test data indicates that each of the 20 MSSVs has a lift distribution or "signature" that should be established before adjustments of the setpoints are made. The valve signatures indicated that, based on a statistical distribution, not all of the valves are capable of lifting within ±1 percent of their setpoint.

This situation was unavoidable and could not be anticipated because a more comprehensive understanding of the overall setpoint variability was only possible following the extensive testing recently performed on the Units 1 and 2 MSSVs. Accordingly, pursuant to 10 CFR 50.91(a)(5), the staff has determined that there are emergency circumstances warranting prompt approval of the proposed change in that failure to act in a timely way will prevent startup of Unit 2.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. It does not involve a significant hazards consideration because:

(1) Analyses demonstrate that the main steam safety valves (MSSVs) with the revised setpoint tolerances will continue to mitigate transients by preventing overpressurization of the reactor coolant system and the main steam system.

In addition, the proposed setpoint revision continues to provide assurance that the steam generator (SG) will not overfill during a SG tube rupture (SGTR) if the lowest set MSSV lifts at the low end of its tolerance band.

(2) There is no physical alteration to any plant system, nor is there a change in the method in which any safety related system performs its function. Any main steam safety valve lifting at the extremes of the proposed tolerance will not result in a low lift setpoint that is less than the normal "no load" system pressure, or a high lift setpoint that allows main steam system overpressurization. Even if all of the MSSVs are at the high end of their tolerance band, overpressurization is precluded.

(3) With the increased MSSV setpoint tolerances, the main steam line safety valves will still prevent pressure from exceeding 110 percent of design pressure in accordance with the ASME code. The conclusions of the FSAR Update accident analyses are unaffected by the change and remain valid.

Accordingly, the Commission has determined that this amendment involves no significant hazards considerations.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, an attempt was made to notify the California State official of the proposed issuance of the amendment. The State official was not available. The State official will be contacted during the week of October 2, 1995.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 amendment involves 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 made a final no significant hazards consideration finding with respect to this amendment. Accordingly, the amendment meets 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 amendment.

7.0 CONCLUSION

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

Principal Contributors: C. Hammer
C. Liang
K. Thomas

Date: October 1, 1995