July 23, 1993

Docket Nos. 50-275
and 50-323

Mr. Gregory M. Rueger
Nuclear Power Generation, B14A
Pacific Gas and Electric Company
77 Beale Street, Room 1451
P.O. Box 770000
San Francisco, California 94177

Dear Mr. Rueger:

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NOS. M77347 AND M77417) AND UNIT NO. 2 (TAC NOS. M77348 AND M77418)

The Commission has issued the enclosed Amendment No. 81 to Facility Operating License No. DPR-80 and Amendment No. 80 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 March 27, 1991, as supplemented October 5, 1992 and July 20, 1993.

These amendments revise TS 3/4.4.4, "Relief Valves," and 3/4.4.9.3, "Overpressure Protection Systems," and their associated bases to implement the recommendations of Generic Letter 90-06. These amendments include additional provisions for power operated relief valve (PORV) and block valve reliability and low temperature overpressure protection.

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

Sincerely,
Original signed by
Sheri R. Peterson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 81 to DPR-80
2. Amendment No. 80 to DPR-82
3. Safety Evaluation

cc w/enclosures:
See next page

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JRoe	PDV r/f
TQuay	OGC
CGrimes	ACRS (10)
Region V (8)	SPeterson
	KPerkins, RV

* See previous concurrence

OFC	LA/PDV	PM/PDV	*OGC	D/PDV
NAME	DFoster	SPeterson:lh	RBachmann	TQuay
DATE	7/22/93	7/22/93	06/16/93	7/23/93

OFFICIAL RECORD COPY DOCUMENT NAME: DC7734

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and 50-323

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OFC	LA/PDV	PM/PDV	OGC	D/PDV
NAME	DFoster	SPeterson:lh	RBachmann	TQuay
DATE	5/14/93	5/26/93	6/16/93	7/23/93

OFFICIAL RECORD COPY DOCUMENT NAME: DC77347.AMD

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

July 23, 1993

Docket Nos. 50-275
and 50-323

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Pacific Gas and Electric Company
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Office of Nuclear Reactor Regulation

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cc w/enclosures:
See next page

Mr. Gregory M. Rueger
Pacific Gas and Electric Company

Diablo Canyon

cc:
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c/o U.S. Nuclear Regulatory Commission
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Avila Beach, California 93424

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. 81
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 March 27, 1991, as supplemented October 5, 1992 and July 20, 1993, 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. 81 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 23, 1993

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. 80
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 March 27, 1991, as supplemented October 5, 1992 and July 20, 1993, 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. 80 , are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of 30 days of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 23, 1993

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

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

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OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. Two Class 1 power-operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.07 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 323°F, MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned.

ACTION:

a. With one Class 1 PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within the next 8 hours.

b. With one Class 1 PORV inoperable in MODES 5 or 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned, restore the inoperable PORV to operable status within 24 hours or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches within the next 8 hours.

c. With both PORVs inoperable, depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within 8 hours.

d. In the event either the PORVs or the RCS vent are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. For Unit 1, Cycle 6:

Operating the PORV through one complete cycle of full travel, and

For Unit 1, Cycle 7 and after, and Unit 2, Cycle 6 and after:

Operating the PORV through one complete cycle of full travel during MODES 3 or 4 with the block valves closed, and

b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 In addition to the requirements of Specification 4.0.5, each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The safety-related nitrogen supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Isolating and venting the normal air supply, and

b. Verifying that any leakage of the Class 1 Backup Nitrogen System is within its limits, and

c. Operating the PORVs through one complete cycle of full travel.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet and two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups 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 the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring heater group power at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations and the unavailability of the steam generator as a heat removing component require that at least two RHR trains be OPERABLE.

The operation of one reactor coolant pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 323°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of the valve's Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown.

RELIEF VALVES (Continued)

protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

Each PORV has a remotely operated block valve to provide a positive shut-off capability should a relief valve become inoperable. Operation with the block valves opened is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The OPERABILITY of the PORVs and block valves in MODES 1, 2, and 3 is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,

2. Manual control of PORVs to control RCS pressure as required for accident mitigation,

3. Manual closing of a block valve to isolate a stuck open PORV,

4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and

5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for accident mitigation.

The non-Class 1 PORV and block valve are used only as a backup to the two redundant Class 1 PORVs and block valves to control RCS pressure for accident mitigation. Therefore continued operation with the non-Class 1 PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their safety functions. Surveillance Requirement 4.4.4.1 addresses the PORVs and Surveillance Requirement 4.4.4.2 addresses the block valves.

Surveillance Requirement 4.4.4.1.a provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS.

Surveillance Requirement 4.4.4.1.b provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for the PORV to be OPERABLE in MODES 1, 2, and 3.

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operation limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of both Class 1 PORVs or an RCS vent opening of at least 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 323°F. Either Class 1 PORV has adequate relieving capability to protect the RCS from overpressurization for all anticipated transients.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 80 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

On June 25, 1990, the staff issued Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." The generic letter represented the technical resolution of the above mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff position and improvement to the plant's technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock and Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating MODEs 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letter dated December 21, 1990, Pacific Gas and Electric Company (or the licensee) responded to GL 90-06. By letter of March 27, 1991, as supplemented October 5, 1992 and July 20, 1993, the licensee submitted a request for changes to the Technical Specifications (TS) in response to GL 90-06. The October 5, 1992 and July 20, 1993 submittals provided clarifying information that did not change the initial proposed no significant hazards determination or expand the scope of the original Federal Register notice.

2.1 EVALUATION FOR GENERIC ISSUE 70

The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70--Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants."

The TS changes in response to Generic Issue 70 consist of the following changes to TS 3/4.4.4, "Relief Valves:"

1. The action required with one or more power-operated relief valves (PORVs) inoperable due to excessive seat leakage would be revised to require that power is maintained to the associated closed block valves(s). The current Action Statement does not require power to be maintained to the closed block valve(s).

2. The shutdown requirements would be revised to be consistent with the mode applicability requirements.

3. The action required with one or more block valves inoperable would be revised to delete the option of closing the inoperable block valve and removing power from it. This change would eliminate the option for continued operation with a Class 1 PORV block valve inoperable.

4. New Surveillance Requirements would be added for the Backup Air/Nitrogen system for the Class 1 PORVs.

5. The block valve Surveillance Requirement would be modified to require operating the valve through one complete cycle of full travel to demonstrate operability when the block valve is closed due to excessive seat leakage of a PORV.

6. Surveillance Requirement 4.4.4.3 has been added to verify the capability of the valves in the safety-related PORV control air systems.

7. Surveillance Requirement 4.4.4.1.a has been revised to require the 18-month PORV stroke test to be performed during Mode 3 or Mode 4. Additionally, the GL requests that PORVs be stroked in all cases prior to establishing conditions where the PORVs are used for low-temperature overpressure protection. This could be interpreted to mean that PORVs should be stroke tested during every shutdown and again during every startup. However, the inclusion of the PORVs in the IST program (also recommended) requires that the valves be tested no more frequently than every three months (unless valve maintenance is performed) to demonstrate operability. The intent of the Generic Letter was not to increase the frequency of PORV stroke testing.

8. The TS Bases would be revised to reflect the proposed changes and to better define the basis for operability of the PORVs and block valves.

The staff has reviewed the licensee's proposed modifications to the Diablo Canyon, Units 1 and 2 Technical Specifications and the associated bases. Since the proposed modifications are consistent with the staff's position previously stated in the GL and found to be justified in NUREG-1316, the staff finds the proposed changes to be acceptable.

2.2 EVALUATION OF GENERIC ISSUE 94

The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

The TS changes in response to Generic Issue 94 consist of the following changes to TS 3/4.4.9.3, "Overpressure Protection Systems:"

1. A new Action Statement would be added that reduces the allowed outage time from 7 days to 24 hours for an inoperable PORV in Modes 5 or 6.

2. An Analog Channel Operational Test would be required to demonstrate PORV operability instead of a Channel Functional Test.

3. The frequency for PORV surveillance would be changed from "within 31 days prior to entering a condition in which the PORV is required Operable" to "at least once per 31 days."

4. The TS and Bases would be revised to identify that the TS only applies to the Class 1 PORVs.

The staff has reviewed the licensee's proposed modifications to the Diablo Canyon, Units 1 and 2 Technical Specifications and the associated bases. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter and justified in the above-mentioned regulatory analysis, the staff finds the proposed modifications to be acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in

10 CFR Part 20 and change 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 (56 FR 24214). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

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