September 18, 1975

SUPPLEMENT NO. 3

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

B/57

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APPENDICES

1.0 INTRODUCTION

The Commission's Safety Evaluation Report in the matter of the Diablo Canyon Nuclear Power Station, Units 1 and 2, was issued on October 16, 1974. In the Safety Evaluation Report it was stated that supplemental reports would be issued to update the Safety Evaluation Report in those areas where the staff's evaluations had not been completed. Supplements 1 and 2 to the Safety Evaluation Report, issued on January 31, 1975 and May 9, 1975, respectively, documented the resolution of several outstanding items, and summarized the status of the remaining outstanding items.

The purpose of this Supplement Number 3 is to further update the Safety Evaluation Report by providing the staff's evaluation of certain matters which were not resolved when Supplement Number 2 was issued. Each of the following sections of this supplement is numbered the same as the corresponding sections of the Safety Evaluation Report that are being updated.

Appendix A of this supplement is a continuation of the chronology of the Nuclear Regulatory Commission staff's principal actions with respect to radiological matters related to the processing of the application. Appendix B is a report by the Advisory Committee on Reactor Safeguards.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.5 Missile Protection Criteria

In a letter to the applicant dated July 11, 1975 we requested additional information describing the feasibility and cost of making improvements to the tornado missile protection for three areas of the plant which are still being reviewed. These areas are as follows:

(1) The 4.16 kV switchgear rooms in the turbine building at elevation 119 do not appear to have substantial protection from tornado missiles which might penetrate the turbine building walls in the vicinity of these rooms. From certain directions, the existing protection consists of relatively light weight exterior wall panels.

(2) The diesel generators and their radiators appear to be vulnerable to small missiles which might go through the spaces between the I - beams in the missile protection shield for the air intakes.

(3) It is not clear what missile resisting capability is available for the pipes which connect to the component cooling surge tanks which are placed on top of the auxiliary building. Also, it is not clear whether the component cooling water system could be operated if damage to the surge tanks or piping were sustained.

In Amendment 33 to the FSAR, the applicant provided (1) a revised design for the diesel generator air intake missile shield, and (2) a description of the consequences of tornado missile damage to the component cooling surge tanks and connected piping. The applicant is

expected to submit additional information regarding the protection for the switchgear rooms. We will report the results of our evaluation of all of this information in a future supplement to the Safety Evaluation Report.

4.0 REACTOR

4.2 Mechanical Design

4.2.1 Fuel

In the Safety Evaluation Report and in Supplement Number 2 to the Safety Evaluation Report, we stated that Westinghouse would document the justification for applying the results of certain tests made on 17 x 17 fuel assemblies with seven spacer grids to 17 x 17 fuel assemblies with eight spacer grids.

The tests in question, which were performed with 17 x 17 fuel assemblies with seven spacer grids, were reported in WCAP-8278, "Hydraulic Flow Test of the 17 x 17 Fuel Assembly", and WCAP-8236, "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant-Accident". We have previously concluded, as described in the Safety Evaluation Report, that these reports are acceptable for seven grid fuel assemblies.

In a letter dated May 20, 1975, Westinghouse documented the justification for applying the results of the seven grid tests to 17 x 17 fuel assemblies with eight spacer grids. Our primary concern with these tests was that the fuel assembly design be able to withstand flow-induced vibration. The information submitted by Westinghouse provides justification for demonstrating that the design of the 17 x 17 fuel assembly is adequate to withstand the effects of flow-induced vibration under both normal operating and transient conditions. We have evaluated this justification and found it to be acceptable. Based on our evaluation of the additional information which was provided, we consider this matter to be resolved.

Based on our evaluation, as described in the Safety Evaluation Report, Supplement Number 2, and this report, we have concluded that the mechanical design of the Diablo Canyon fuel is acceptable.

4.4 Thermal and Hydraulic Design

In the Safety Evaluation Report (see page 4-15) we stated that the facilities were designed to operate at a higher heat output and temperature than comparable units. This conclusion was based on the use of the THINC code which permitted a more detailed analysis of the thermal and hydraulic characteristics of the core. However, we required verification of the calculational accuracy of the code, based on the Zion facility. In Supplement Number 2 to the Safety Evaluation Report, we stated that the results of the verification test program for the THINC code had been documented in topical reports WCAP-8453 (Proprietary) and WCAP-8454 (Non-Proprietary), "Analysis of Data from the Zion (Unit 1) THINC Verification Test," December 1974. Since our evaluation of these reports was not completed at that time, we further stated that, if sufficient verification could not be obtained from the combined test and analytical programs, restrictions would be included in the Diablo Canyon technical specifications to maintain required margins to fuel rod damage.

We have now completed our evaluation of the above mentioned topical reports; this evaluation is documented in a letter to Westinghouse dated July 9, 1975. Based on our evaluation, we have concluded that no further in-core testing is required for the THINC code, since the THINC calculations gave conservative results with

respect to the Zion test results. Because the Zion and Diablo Canyon cores have similar thermal and hydraulic characteristics, we have concluded that no special operating restrictions will be required for the Diablo Canyon Units. We consider this matter to be resolved.

In Supplement Number 2 to the Safety Evaluation Report, we stated that we considered the effects of fuel rod bowing on departure from nucleate boiling heat flux after the first fuel cycle to be unresolved, and that we would report the resolution of this matter in a future supplement to the Safety Evaluation Report. Westinghouse has submitted additional information concerning fuel rod bowing after the first fuel cycle.

We have not completed our evaluation of this information, and we expect additional data to be submitted. We will report the resolution of this item in a future supplement to the Safety Evaluation Report.

In Supplement Number 2 to the Safety Evaluation Report, we stated that, if the results of the non-uniform departure from nucleate boiling tests were not available when the technical specifications for Diablo Canyon were finalized, we would require that the minimum allowable departure from nucleate boiling ratio be increased 5 percent above that required to satisfy the 95/95 criterion (see page 4-12 of the Safety Evaluation Report). These results have been documented in WCAP-8536 (Proprietary), "Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometries with 22 Inch Grid Spacing."

We have not completed our evaluation of this information, and will report the resolution of this item in a future supplement to the Safety Evaluation Report.

Based on our evaluation, as described in the Safety Evalution Report, Supplement Number 2, and this report, we have concluded, subject to favorable resolution of the outstanding items described above, that the thermal and hydraulic design of the Diablo Canyon reactors is acceptable, and that these reactors can operate at the proposed core power levels.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

Regarding the blowdown forces acting on the reactor pressure vessel in the unlikely event of a loss-of-coolant accident, we were informed on May 7, 1975, by a Licensee of a pressurized water reactor, Virginia Electric and Power Company, that an asymmetric loading resulting from a postulated pipe rupture at a particular location in the reactor coolant loop had not been taken into account in the original design analysis of the reactor vessel support system. This loading results from the forces induced on the internals within the reactor vessel during such a loss-of-coolant accident caused by transient differential pressure conditions within the vessel. In addition, the asymmetric loading from the transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture was not included in the original design analysis. However, the symmetric loadings from such a pipe rupture were included in the original analysis of the reactor vessel support system.

It is our opinion that these factors related to the adequacy of vessel support systems are generic in nature, and apply to the Diablo Canyon Units. It is also our opinion that the original design of the reactor vessel support system for these Units was developed using conservative methods of analysis for the symmetric loads that were considered. Accordingly, it is likely that, when a systematic estimate of the conservatism used in the original analysis is completed, and the asymmetric

loads are properly taken into account, it will be confirmed that, although the margins of safety will be less than previously thought, the vessel support system can safely withstand the postulated effects of this particular loss-of-coolant accident.

We are continuing our review of this matter, and will report the results of our evaluation in a future supplement to the Safety Evaluation Report.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

With regard to subcompartment pressure calculations, in Supplement Number 2 to the Safety Evaluation Report we stated that the applicant must provide additional information on the geometry of the system to justify the assumed limitation on the size of the opening that can result from a break at the reactor vessel nozzle weld. We further stated that, with regard to the pressure response of the reactor coolant system pipe penetration through the reactor shield structure, the applicant must analyze the response of the piping penetration to a pipe break within the penetration using an acceptable method of analysis, or else justify that a break need not be postulated inside the penetration.

In Amendment 30 to the FSAR, the applicant provided additional information regarding (1) the limited displacement rupture of the reactor coolant pipe at the nozzle weld, and (2) justification that a pipe break need not be assumed inside the piping penetration. The applicant demonstrated that the geometrical configuration limits the size of the opening that can result from a break at the nozzle weld. In addition, the applicant has used the appropriate guidelines in Regulatory Guide 1.46 to justify that a break need not be postulated inside the piping penetration.

We have reviewed the information submitted and, based on our evaluation, we have concluded that the applicant has adopted acceptable criteria for postulated break locations and type, and has provided an acceptable basis for the assumed limited displacement of the ruptured pipe. We consider these matters regarding subcompartment pressure calculations to be resolved.

6.2.2 Containment Heat Removal Systems

Because of the possibility that insulation could be torn from piping or equipment inside containment during a loss-of-coolant accident and block the flow of water through the sump screens, we asked the applicant to perform an evaluation to determine what amounts of insulation might become detached due to jet effects created by a pipe break inside containment. In Amendment 29 to the FSAR the applicant provided information regarding the design of piping and equipment insulation used within the containment, and the expected behavior of this insulation during a postulated loss-of-coolant accident.

Both reflective metal panel insulation and stainless steel jacketed calcium silicate insulation are used inside containment. Jet effects could dislodge some of this insulating material from piping and equipment. However, the arrangement of floors, walls, and compartments would prevent some portion of this dislodged insulation from reaching the containment sump.

To prevent blockage of the recirculation flow at the sump, the design of the Diablo Canyon containment sump includes a debris curb, trash rack, double screens and large screen areas which cause very low flow velocities as the water approaches and passes through the screens. This permits most of the debris to settle out before reaching the containment sump. The applicant has also performed a jet force analysis to estimate the maximum amount of piping and equipment insulation which could be removed in the vicinity of a pipe rupture. The analysis shows that the total surface area of insulation removed is significantly less than the sump coarse screen area.

Based on our evaluation of the information submitted by the applicant, we have concluded that the use of reflective metal insulation and steel jacketing around the conventional insulation, coupled with the design features of the containment sump, provide adequate assurance that the sump will not be blocked by displaced insulation during a loss-of-coolant accident. We consider this matter to be resolved.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Design Bases

In Supplement Number 2 to the Safety Evaluation Report, we stated that the applicant had provided a commitment to lock out power to those motor-operated valves identified by the staff as affecting the function of the ECCS if spurious operation were to occur. The valves for which locking out of power would be required were those identified in item 7.12 of a request for additional information (see items 42 and 45 in Appendix A to the Safety Evaluation Report). At the time we considered this matter to be resolved.

Since that time, two additional valves, the containment sump isolation valves (Valve Numbers 8982, A and B), have been identified as a potential problem where spurious operation could result in the loss of system function. These valves are located in two lines from the sump, and are required to operate for realignment to the recirculation mode. Each line communicates with one of the two residual heat removal pump suction lines. These in turn connect to the refueling water storage tank through a common line which contains a check

valve. Each containment sump isolation valve is interlocked with a motor operated valve in its individual pump suction line to prevent communication between the containment sump and the common line to the refueling water storage tank. However, in the event of an electrical failure causing spurious mechanical motion to open either sump isolation valve during the early safety injection phase following a loss-of-coolant accident, containment pressure (up to 47 psig) could force closure of the check valve in the common line to the refueling water storage tank. This could result in cavitation and damage to both residual heat removal pumps, one of which is required for long-term cooling following a loss-of-coolant accident.

We have discussed this matter with the applicant, and requested additional information regarding our concerns with this matter. In Amendment 33 to the FSAR, the applicant proposed to lock out power to the containment sump isolation valves mentioned above (Valve Numbers 8982, A and B). We are evaluating this information submitted by the applicant, and will report the final resolution of this item in a future supplement to the Safety Evaluation Report.

6.3.3 Performance Evaluation

In Supplement Number 2 to the Safety Evaluation Report, we stated that the applicant would submit a revised analysis of the emergency core cooling system performance.

In Amendment 33 to the FSAR the applicant submitted a revised analysis. However, the analysis submitted was valid only for Diablo Canyon Unit 1, and it did not contain all of the information which we had previously requested (see item 136 in Appendix A of this report). We will review the analysis in Amendment 33 as well as the additional information to be submitted by the applicant, and will report the results of our review in a future supplement to the Safety Evaluation Report prior to a decision concerning the issuance of operating licenses for Diablo Canyon Units 1 and 2.

6.3.5 Conclusion

Our evaluation of the acceptability of the emergency core cooling system has not been completed. The items which are currently unresolved

are described in Sections 6.3.1 and 6.3.3 of this report. We will report our conclusions regarding the acceptability of the emergency core cooling system in a future supplement to the Safety Evaluation Report.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features

In June of this year (see item 135 in Appendix A of this report), we asked the applicant to provide information regarding the potential effects of secondary system fluid flow instability. Events such as damage to the feedwater system piping that occurred at Indian Point 2 on November 13, 1973, could originate as a consequence of the uncovering of the feedwater sparger in the steam generator or uncovering of the steam generator feedwater or auxiliary feedwater inlet nozzles.

We have asked the applicant to provide information which demonstrates that unacceptable damage will not result from feedwater hammer on the Diablo Canyon Units. This information can be provided in the form of adequate testing which would be performed in accordance with plant procedures.

The applicant provided the information on feedwater hammer in August of this year (see item 142 in Appendix A of this report). We are currently evaluating this information, and will report the results of our evaluation in a future supplement to the Safety Evaluation Report. This matter must be resolved before the plant can be operated at power.

15.0 ACCIDENT ANALYSES

15.1 General

In Supplement Number 2 to the Safety Evaluation Report, we stated that the applicant must justify the assumption that air entering the auxiliary building would be heated 15 degrees Fahrenheit before it reaches the auxiliary building charcoal filters. This was the only outstanding issue remaining with regard to dose reduction in the event of residual heat removal leakage following a postulated loss-of-coolant accident.

In Amendment 28 to the FSAR, the applicant provided additional information with regard to this matter. The applicant has described the heat loads used to calculate the 15 degree temperature rise. We have reviewed this information and found it to be acceptable.

Based on our evaluation of the applicant's calculations, we have concluded that adequate humidity control has been provided for the charcoal filters in the event of residual heat removal system leakage following a postulated loss-of-coolant accident. We consider this matter to be resolved.

17.0 QUALITY ASSURANCE

17.3 Quality Assurance Program

In Section 22 of Supplement Number 2 to the Safety Evaluation Report, we stated that we had not completed our evaluation of the applicant's commitment to the guidance in certain WASH documents which pertain to the operational quality assurance program.

In Amendment 30 to the FSAR the applicant revised this commitment. The applicant stated that the quality assurance program for plant operation of Diablo Canyon, Units 1 and 2, complies with the guidance contained in WASH-1283 (May 24, 1974), "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants - Revision 1"; WASH-1284 (October 26, 1973), "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants"; and WASH-1309 (May 10, 1974), "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants." This complies with our position on the implementation of guidance in quality assurance programs and is, therefore, acceptable. We consider this matter to be resolved.

Based on our evaluation as described in the Safety Evaluation Report and supplemented in this report, we now conclude that the Diablo Canyon quality assurance program has the necessary controls to comply with the requirements of Appendix B to 10 CFR Part 50 and is, therefore, acceptable for controlling the operational phase of Diablo Canyon, Units 1 and 2.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The Advisory Committee on Reactor Safeguards completed a partial review of the application for operating licenses for Diablo Canyon, Units 1 and 2, at its 182nd meeting, June 5-7, 1975. The Diablo Canyon Units were previously considered at Subcommittee meetings held in Washington, D.C. on September 12, 1974, in San Luis Obispo, California on February 18-19, 1975; and in Los Angeles, California on May 23, 1975. A copy of the Committee's report on this partial review, dated June 12, 1975, is attached as Appendix B to this report.

We will report the actions we have taken or plan to take in response to the Committee's comments in a future supplement to the Safety Evaluation Report.

22.0 CONCLUSIONS

In Section 22 of Supplement Number 2 to the Safety Evaluation Report, we stated that several items were still outstanding, and that favorable resolution of these items would be required before operating licenses for Diablo Canyon Units 1 and 2 could be issued. A number of these items have been resolved in this supplement. A revised status report is given below. It includes those items which were not resolved in Supplement Number 2 (items1 through 14 below), as well as new unresolved items which have arisen since Supplement Number 2 was issued (items 15 through 19 below).

(1) The applicant has provided additional information on the effects of tsunamis caused by near-shore generators; however, our evaluation of this information has not been completed (Sections 2.4.2, 2.4.3 and 2.4.5 of Supplement Number 1).

(2) Our comparative evaluation of the Hosgri and Santa Lucia Bank faults, and our evaluation of the earthquake potential of the Hosgri Fault have not been completed. The applicant must provide additional information before our evaluation can be completed (Sections 2.5.1 and 2.5.2 of Supplement Number 1).

(3) The applicant has provided information on the potential consequences of pipe breaks outside containment. During our evaluation of this information we have determined that additional information must be provided (see Item 143 in Appendix A of this report). (Section 3.6 of the Safety Evaluation Report).

(4) The applicant has submitted some of the required information concerning the seismic qualification of Category I instrumentation

and electrical equipment. Additional information must be submitted before our evaluation can be completed (Sections 3.10 and 7.8 of the Safety Evaluation Report).

(5) Our evaluation of the justification for applying the results of tests of 7-grid assemblies to 8-grid assemblies has been completed, and this matter is resolved (Section 4.2.1 of this report).

(6) The matters regarding uncertainties in the thermal and hydraulic design have been partially resolved (Section 4.4 of this report).

(7) We have completed our evaluation of the matters regarding subcompartment pressure calculations using the Transient Mass Distribution Program. This matter is resolved (Section 6.2.1 of this report).

(8) We have not completed our evaluation with regard to single failures in the containment sump isolation valves (Sections 6.3.1 and 6.3.5 of this report).

(9) The applicant has submitted part of the required emergency core cooling system analysis in accordance with the Final Acceptance Criteria. Additional information must be submitted before our evaluation can be completed (Sections 6.3.3 and 6.3.5 of this report).

(10) The applicant has provided information regarding physical separation in the process analog system. Additional information must be provided before our evaluation can be completed (Section 7.2.3 of the Safety Evaluation Report).

(11) Our evaluation of the Westinghouse generic ATWS model has not been completed (Section 7.2.5 of the Safety Evaluation Report).

(12) The applicant has provided some information concerning the environmental qualification of Category I instrumentation and electrical equipment. Additional information must be submitted before our evaluation can be completed (Section 7.8 of the Safety Evaluation Report).

(13) Our evaluation of the design modifications to the auxiliary building to bring about a reduction of the doses in the event of residual heat removal system leakage following a postulated loss-of-coolant accident has been completed. This matter is resolved (Section 15.1 of this report).

(14) The matter regarding the operational quality assurance program is resolved (Section 17.3 of this report).

(15) Our evaluation of the potential effects of insulation used inside containment on sump operation following a loss-of-coolant accident has been completed. This matter is resolved (Section 6.2.2 of this report).

(16) The applicant has submitted information concerning the potential for damage due to feedwater system water hammer. Our evaluation of this information has not been completed (Section 10.4 of this report).

(17) The applicant has submitted additional information regarding tornado missile protection. Our evaluation of this information has not been completed, and further information must be provided (Section 3.5 of this report).

(18) Regarding fire protection in the vicinity of electrical cabling and equipment, the applicant has not provided the additional information that was requested (Item 143 in Appendix A of this report).

(19) Our evaluation of the factors related to the adequacy of the reactor vessel support systems has not been completed (Section 5.2.1 of this report).

Subject to favorable resolution of the outstanding matters described above, the conclusions as stated in Section 22 of the Safety Evaluation Report remain unchanged.

APPENDIX A

CONTINUATION OF THE CHRONOLOGY OF THE RADIOLOGICAL REVIEW

127.	May 9, 1975	Supplement No. 2 to the Safety Evaluation Report issued.
128.	May 16, 1975	Submittal of Amendment No. 28 consisting of additional information required for the resolution of outstanding items in the Safety Evaluation Report.
129.	May 19, 1975	Letter to applicant transmitting the schedule for implementation of the Westinghouse Standard Technical Specifications.
130.	May 23, 1975	ACRS Subcommittee Meeting in Los Angeles, California.
131.	June 2, 1975	Submittal of Amendment No. 29 consisting of additional information required for the resolution of outstanding items in the Safety Evaluation Report.
132.	June 5, 1975	ACRS Full Committee Meeting in Washington, D.C.
133.	June 12, 1975	ACRS letter which constitutes a partial review of the Diablo Canyon operating license application.
134.	June 13, 1975	Letter to applicant requesting additional information on boron precipitation effects on long term cooling for the emergency core cooling system.
135.	June 13, 1975	Letter to applicant requesting additional information on the effects of secondary system fluid flow instabilities.
136.	July 9, 1975	Letter to applicant requesting additional information on the emergency core cooling system.

137.	July 11, 1975	Letter to applicant requesting additional information on tornado missile protection for the Diablo Canyon Units.
138.	July 16, 1975	Letter to applicant requesting additional information regarding the extension of the construction permits for Units 1 and 2.
139.	July 29, 1975	Submittal of report on the consequences of seismic-induced actuation of protection system relays on the Diablo Canyon Units.
140.	July 29, 1975	Submittal of Amendment No. 30 consisting of additional information required for the resolution of outstanding items in the Safety Evaluation Report.
141.	July 31, 1975	Meeting with applicant to discuss geology and seismology.
142.	August 5, 1975	Letter from applicant transmitting the information on the effects of secondary system fluid flow instabilities that was requested in our letter of June 13, 1975.
143.	August 7, 1975	Letter to applicant requesting additional information on electrical instrumentation and control systems, pipe break outside containment, and fire protection.
144.	August 15, 1975	Submittal of Amendment No. 31 consisting of a partial response to the staff's request for additional information on geology and seismology dated February 12, 1975.
145.	August 19, 1975	Submittal of Amendment No. 32 consisting of a partial response to the staff's request for additional information on geology and seismology dated February 12, 1975.
146.	August 20, 1975	Submittal of a marked-up copy of the Westinghouse Standard Technical Specifications for the Diablo Canyon Units.

147.	August 28, 1975	Meeting with applicant to discuss geology and seismology.
148.	August 29, 1975	Submittal of Amendment No. 33 consisting primarily of a revised emergency core cooling system analysis.
149.	September 4, 1975	Letter to applicant transmitting our review of the Westinghouse Protection System Noise Test Report, and requesting additional information on this item.
150.	September 16, 1975	Submittal of Amendment No. 34 consisting of a partial response to the staff's request for additional information on geology and seismology dated February 12, 1975.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 12 1975

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: REPORT ON PARTIAL REVIEW OF DIABLO CANYON NUCLEAR POWER STATION UNITS 1 AND 2

Dear Mr. Anders:

At its 182nd meeting, June 5-7, 1975, the Advisory Committee on Reactor Safeguards completed a partial review of the application of the Pacific Gas and Electric Company for authorization to operate the Diablo Canyon Nuclear Power Station Units 1 and 2. The project was previously considered at Subcommittee meetings in Washington D.C. on September 12, 1974; in San Luis Obispo, California on February 18-19, 1975; and in Los Angeles, California on May 23, 1975. During its review, the Committee had the benefit of discussions with representatives and consultants of the Pacific Gas and Electric Company, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for the Diablo Canyon Nuclear Power Station Unit 1 in its letter of December 20, 1967, and for Unit 2 in its letter of October 16, 1969.

The site is located on 750 acres adjacent to the Pacific Ocean in San Luis Obispo, County, and is approximately 12 miles west-southwest of the city of San Luis Obiso.

The two units at the Diablo Canyon Station are essentially identical. Each includes a four-loop Westinghouse nuclear steam supply system similar in most respects to that for the Trojan Nuclear Plant, on which the ACRS reported on November 20, 1974. The design core power level for Unit 1 is 3338 MW(t) and Unit 2 is 3411 MW(t). The slight difference in output for the two units is due to the upgraded turbine generator design for Unit 2.

The Committee has not completed its review of the seismic design bases, the adequacy of the seismic design, or of the requirements with regard to protection against tsunamis. These, and some additional matters discussed below, will be reviewed by the Committee following completion of review of seismic-related topics by the NRC Staff.

The Diablo Canyon Units 1 and 2 are scheduled to be among the first to go into operation using a full-core of 17x17 fuel. While many of the various required verification programs have been completed and reviewed by the NRC Staff, other tests and analyses are still to be documented and reviewed. These include: DNB tests for non-uniform heat flux, fuel assembly flow tests, and the effect of fuel rod bowing on DNB after the first fuel cycle. The results of such tests and analyses should be evaluated fully by the NRC Staff, and resolved to it's satisfaction, prior to the full-core use of 17x17 fuel to produce power. Prototype 17x17 fuel rod assemblies are to be loaded into operating pressurized water reactors in the near future; the results of these irradiations should be followed closely. The Committee wishes to be kept informed concerning the results of the various ongoing 17x17 test and analytical programs, and any design changes which may be proposed in the future.

Following each cycle of operation, 17x17 fuel assemblies will be examined for fuel rod integrity, fuel rod and assembly dimension and alignment, and surface deposits. In view of the fact that the 17x17 fuel array is a new design and that no prototype irradiations are planned for 17x17 fuel containing eight spacer-grids, the results of surveillance programs for this type fuel should be followed closely. The Committee wishes to be kept informed.

The recently proposed method of constant axial offset control will be used for core power distribution monitoring and control. The NRC Staff should review the effectiveness of this method in protecting against adverse consequences of postulated reactor transients and accidents. The Committee wishes to be kept informed.

Several changes have been made in the Westinghouse ECCS evaluation model to bring it into conformance with the Commission criteria as given in 10 CFR 50, Appendix K. The performance of the emergency core cooling systems will be reevaluated with the approved evaluation model, and appropriate operating limits and procedures for ensuring monitoring of the power distribution are to be incorporated in the Technical Specifications. The Committee wishes to be kept informed.

The evaluation of Anticipated Transients Without Scram has been made generically for Westinghouse plants, and the applicant has made comparisons indicating that the results obtained are applicable to Diablo Canyon Units 1 and 2. NRC Staff review should be completed and this matter resolved in a manner satisfactory to the NRC Staff and the ACRS.

Diablo Canyon Units 1 and 2 may be among the first reactors of this type to operate at a power as high as 3411 MW(t). Because there is limited operating experience with very large, high-power density reactors, the ACRS has previously recommended a more cautious-than-normal approach to full power, with longer periods of operation at power levels in the range of 70 to 90% of full power, and with additional monitoring of core and systems performance throughout the life of the first core. The applicant discussed with the Committee an augmented startup program, which is proposed for implementation on several of the first plants to operate with a full-core employment of the 17x17 fuel assembly. The Committee believes that the augmented program is desirable and recommends that the NRC Staff evaluate the results of this program, as well as overall operating experience with large high power-density reactors, prior to sustained operation at full power.

Certain aspects of the protection against tornadic missiles are still under evaluation. This matter should be resolved in a manner satisfactory to the NRC Staff.

The applicant has not provided adequate information to confirm the environmental qualification of Class I instrumentation and electrical equipment. This matter should be resolved in a manner satisfactory to the NRC Staff and the ACRS.

Generic problems relating to large water reactors have been identified by the NRC Staff and the ACRS and discussed in the Committee's report dated March 12, 1975. These problems should be dealt with appropriately by the NRC Staff and the applicant as suitable approaches are developed.

Several unresolved items were identified by the NRC Staff in their Supplement No. 2 to the Safety Evaluation Report, and at the May 23, 1975 Subcommittee Meeting. The ACRS expects these to be resolved in timely fashion and plans to review several of them, including matters relating to water-hammer effects and subcompartment pressures in a postulated LOCA, in connection with its further review of seismic-related aspects. There also remain some systems behavior and interactions questions and some questions concerning forces on the pressure vessel support structure during blowdown for certain postulated accidents which the Committee expects to review further.

Excepting the seismic and other matters identified above as requiring further Committee review, the ACRS believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Diablo Canyon Nuclear Power Station Units 1 and 2 can be operated at power levels up to 3338 and 3411 MW(t), respectively, without undue risk to the health and safety of the public. The Committee will report in the future on those matters not reviewed herein.

Sincerely,

W Kerr

W. Kerr
Chairman

References

1. Final Safety Analysis Report (FSAR) for the Diablo Canyon Nuclear Power Station, Units 1 & 2, and Amendments 1-28 to the FSAR.
2. Safety Evaluation Report dated October 16, 1974, and Supplements 1 & 2 dated January 31, 1975 and May 9, 1975, respectively.
3. Letter dated April 7, 1975, Pacific Gas and Electric Company (PG&E) to NRC, concerning evaluation of corrosion resistance of alternate steam generator tube materials.
4. Emergency Plan for the Diablo Canyon facility dated March 21, 1974, and Appendix I dated June 1974.

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Regulatory File Cy

PACIFIC GAS AND ELECTRIC COMPANY

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August 15, 1975

Mr. L. C. Rouse
Chief, Fuel Cycle Licensing Branch 1
Division of Materials and
Fuel Cycle Facility Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Docket 50-275-OL
Diablo Canyon Site
Unit 1

Dear Mr. Rouse:

Your letter dated June 17, 1975, asked that we propose additional steps to assure that new fuel in storage will not achieve criticality, assuming that structural failure of the storage facilities might occur leading to flooding and loss of spacing between fuel assemblies.

In order to achieve criticality, <u>both</u> a loss of spacing between fuel assemblies and flooding of the fuel assemblies must occur. We have reviewed the Diablo Canyon design and have identified potential sources of water, failure modes, and flow paths which could contribute to accidental flooding of the fuel storage facilities. It is our conclusion that such flooding is extremely unlikely, even if a seismic event is postulated which results in a loss of spacing between fuel assemblies.

However, in order to provide further assurance that criticality will not occur, assuming the occurrence of a seismic event resulting in a loss of spacing between fuel assemblies, PG&E proposes to take the following additional steps for storage of the initial core loading of new fuel assemblies for Diablo Canyon Unit 1:

1. Store all new fuel assemblies in the spent fuel storage pool.
2. Store all burnable poisons and control rod clusters in their respective fuel assemblies.
3. Add an isolation valve in the fire system in order to isolate the fire main feeding fire stations at elevation 140 ft. in the fuel handling area.

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DIABLO CANYON UNIT 1

SAFETY EVALUATION

Storage of New Fuel Assemblies
for Initial Core Loading

August 15, 1975

SAFETY EVALUATION

The occurrence of criticality is the only event involving the storage of new fuel at Diablo Canyon which could adversely affect the health and safety of the public. The possibility of this occurrence has been evaluated, assuming that the additional steps proposed by PG&E for storage of new fuel assemblies are taken. The results of this evaluation are as follows:

> Loss of spacing between fuel assemblies could not result in criticality unless it were accompanied by the addition of unborated water to the spent fuel pool. Potential sources of water, failure modes, and flow paths have been identified which could result in water entering the spent fuel pool. It is concluded that there is reasonable assurance that water will not enter the pool in a manner which could reduce boron concentration to a level where criticality could occur.

Criticality Analysis - Effect of Boron Concentration

An analysis has been made to determine the relationship between boron concentration and Keff for new fuel assemblies stored in the spent fuel pool, with loss of spacing between fuel assemblies assumed and with the pool intentionally flooded with borated water. This analysis made use of the "Leopard" computer code and assumed the following:

1. 17 x 17 fuel assemblies, Zr-clad, 12 ft. active fuel length.
2. A surface-to-surface spacing between fuel assemblies of 0.5 in. (A surface-to-surface spacing of 0.3 in. is optimum for criticality, but the presence of fuel racks results in each assembly essentially being contained in a box constructed from 1/4 in. angle.)
3. An infinite array of fuel assemblies, each enriched to 3.1% U-235 by weight.
4. All assemblies completely submerged in water at 68°F and 14.7 psia.
5. No poison effects from the presence of stainless steel fuel racks.
6. No burnable poisons or control rod clusters present.

The results of this analysis show that Keff = 0.999 with 2125 ppm boron. At the proposed boron concentration of 4550 ppm, Keff is less than 0.8. The presence of burnable poisons and control rod clusters, the poisoning effect of the stainless steel fuel racks, and the fact that the array of stored fuel assemblies is finite, all contribute to the conservatism of this analysis.

Potential for Reduction of Boron Concentration

The only mechanism which could lead to a reduction in boron concentration and the possibility of criticality is the addition of sufficient water to the pool in a manner which would reduce the boron concentration, at least locally, to a value less than 2125 ppm. All potential sources of such water have been considered in order to identify any mechanism which might lower boron concentration to this level. The potential sources of water have been categorized as follows:

1. Rainfall

2. The raw water reservoir

3. Storage tanks

4. Piping above elevation 140 ft.

5. Piping below elevation 140 ft.

Rainfall - Rainfall, in combination with a tornado or seismic event which damages the roof over the spent fuel pool, could result in the addition of water to the pool and a reduction in boron concentration. If a loss of fuel assembly spacing is also assumed to occur, the possibility of criticality becomes a consideration.

In the event of a tornado in combination with rainfall, it was assumed that the roof over the spent fuel pool was damaged or removed resulting in all rainfall falling on the roof area entering the spent fuel pool. It was further assumed that 5 ft. of borated water was removed from the spent fuel by the tornado (See G.E. APED-5696). For the probable maximum precipitation given in Section 2.4 of the Diablo Canyon FSAR, 26 hours would be required to fill the spent fuel pool to the deck elevation of 140 ft. Assuming complete mixing, the boron concentration in the pool would still be above 3667 ppm and Keff would be below 0.84. Further rainfall would be expected to overflow from the pool with very little mixing.

Rainfall in combination with a seismic event would have less effect on boron concentration than the combination of rainfall and tornado, since the tornado was assumed to remove the upper 5 ft. of borated water in the spent fuel pool.

Raw Water Reservoir - The possibility of reservoir failure and the potential for flooding has previously been considered and is discussed in the Diablo Canyon FSAR (Sections 2.4.4, 2.5.5, and 9.2.4). Although PG&E believes that reservoir failure as a result of a seismic event is unlikely, such an event has been considered. Because of the drainage paths available to accommodate water, the distance between the reservoir and the spent fuel

pool, and the elevation difference of 25 ft. between the edge of the spent fuel pool and the ground elevation at the east side of the fuel handling building, reservoir failure would not result in water entering the spent fuel pool.

Storage Tanks - There are two storage tanks which are potential flooding hazards to the fuel handling building; the refueling water storage tank, which could contain up to 250,000 gallons of water above elevation 140 ft., and the condensate storage tank, which could contain up to 200,000 gallons of water above elevation 140 ft. Both of these tanks are Design Class I. If it is assumed that a seismic event causes these tanks to fail, failure would be predicted to occur at the points of highest stress near the bottom of the tanks. Such failures would not result in water entering the spent fuel pool.

A failure mode has been postulated which could result in water entering the spent fuel pool. It was assumed that the entire superstructure of the fuel handling building fails as a result of a seismic event, that the main roof support beams collapse or deform sufficiently to allow a tank to bend over at an angle of 30°, and that the tank contacts the concrete deck at elevation 140 ft., and bursts open along a weld seam. It was further assumed that no loss of water other than that spilling from the tank onto the 140 ft. deck results. In the event of such a failure, water would fill the 2½ ft. portion of the spent fuel pool between the normal operating level and the 140 ft. deck and then spill over into the cask decontamination area, over edges of the deck, and down stairways. Mixing in the spent fuel pool would be essentially confined to the upper 10 ft. of the pool. Since more than 27 ft. of borated water covers the top of the fuel, the boron concentration would remain above 4000 ppm and Keff below 0.81.

Piping Above Elevation 140 Ft. - There are two 2 in. fire lines and one 1 in. service water line above elevation 140 ft. whose failure could result in water entering the spent fuel pool. The fire lines are normally pressurized. The service water line is intermittently pressurized. For purposes of this safety evaluation, all three lines were assumed to be pressurized and failure due to a seismic event was assumed. It was further assumed that all water discharged from broken lines into the spent fuel pool. The maximum flow possible from these lines is 880 GPM. If complete mixing is assumed in the spent fuel pool, 4½ hours would be required to reach a boron concentration of 2600 ppm and a Keff = 0.945. Several means are available to isolate these lines and terminate flow into the spent fuel pool. Our proposed addition of another isolation valve in the fire main would make it possible to isolate these two 2 in. fire lines without isolating other portions of the fire system.

The assumption of complete mixing is conservative, since mixing would be confined largely to the upper portion of the pool. In any event, there is more than adequate time available to isolate these lines before boron concentration is reduced sufficiently for criticality to occur.

Piping Below Elevation 140 Ft. - A number of lines carrying water are routed below elevation 140 ft. in the vicinity of the spent fuel pool. These lines are separated from the pool vertically by at least one concrete deck and horizontally by the spent fuel pool wall. Potential failure of these lines has been considered. Storage volumes and drainage paths available and the elevation difference between these lines and the top of the spent fuel pool are more than sufficient to prevent water from reaching elevation 140 ft. and entering the spent fuel pool. The presence of at least one deck between these lines and the 140 ft. elevation prevents water discharging from a broken line from spraying upward and reaching the spent fuel pool.