

May 9, 1975

SUPPLEMENT NO. 2

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

3/68

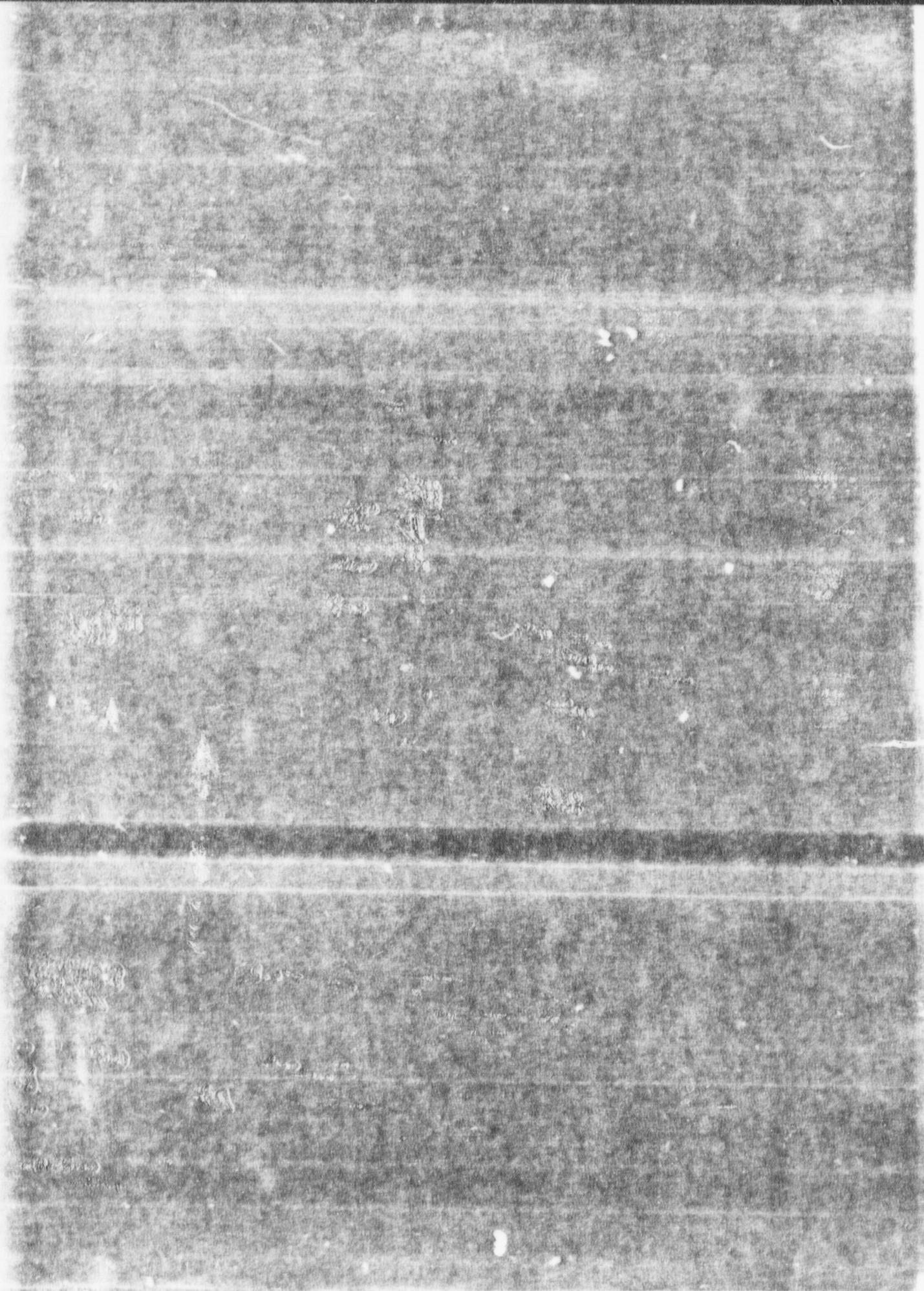


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INTRODUCTION

1.0 The Commission's Safety Evaluation Report (SER) in the matter of the Diablo Canyon Nuclear Power Station, Units 1 and 2 was issued on October 16, 1974. In the SER it was stated that supplemental reports would be issued to update the SER in those areas where the staff's evaluations had not been completed. Supplement No. 1 to the SER, issued on January 31, 1975, documented resolution of several outstanding SER items, and summarized the status of the remaining outstanding items.

The purpose of this Supplement No. 2 is to further update the SER by providing the staff's evaluation of certain matters which were not resolved when Supplement No. 1 was issued. Each of the following sections of this supplement is numbered the same as the sections of the SER that are being updated.

Appendix A of this supplement is a continuation of the chronology of the NRC staff's principal actions with respect to radiological matters related to the processing of the application. Appendix B is the report of our meteorology consultant, the National Oceanic and Atmospheric Administration (NOAA). Appendix C is a listing of errata to the SER and Supplement No. 1.

2.0 SITE CHARACTERISTICS2.3 Meteorology2.3.3 Onsite Meteorological Measurements Program

In the SER, we stated that the applicant would need to provide more detailed information on control room monitoring of meteorological parameters before our evaluation could be completed.

In Amendment 24, the applicant described an operational onsite meteorological program. It includes displays of thirty minute mean values of: wind speed and direction at the 25-ft and 250-ft levels; vertical temperature gradients between the 25-ft and 250-ft levels and between the 25-ft and 150-ft levels; standard deviations of vertical and azimuthal angle fluctuations at the 25-ft and 250-ft levels; and dewpoint temperature at the 25-ft level. The thirty minute mean values will be updated at five minute intervals. The control room display also includes two sets of values for centerline relative concentration (X/Q) and horizontal standard deviation of the plume (σ_y). These values are displayed for distances of 0.8, 1.0, 2.0, 4.0, 6.0, 8.0 and 10.0 kilometers. One set of X/Q values is based upon atmospheric stability defined by vertical temperature gradient. The other set is based on atmospheric stability defined by azimuthal and vertical angle fluctuations. For accident conditions the most conservative set of relative concentration values will be assumed valid.

The determination of horizontal and vertical standard deviations of the plume (σ_y and σ_z) based on the relationships with azimuthal and vertical angle fluctuations (expressed in radians) as described on page 2.3-16c of the FSAR is acceptable. In the FSAR, the applicant did not specify the method of determining the values of σ_y and σ_z based on vertical temperature gradient. We will require that σ_y and σ_z based on vertical temperature gradient be determined from the standard curves in Meteorology and Atomic Energy - 1968,* Appendix A, Figures A.2 and A.3, with σ_z limited to 1000 meters. Curves for type G stability can be derived from the following relationships:

$$(\sigma_y) (G) = \frac{2}{3} (\sigma_y) (F)$$

$$(\sigma_z) (G) = \frac{3}{5} (\sigma_z) (F)$$

We conclude that the parameters and mode of control room display are acceptable. However, the equation for the calculation of X/Q presented on page 2.3-16b of the FSAR, is not acceptable. We will require that the X/Q values on display in the control room be based on the following standard ground-level release model with adjustments for building wake:

$$\frac{X}{Q} = \frac{1}{U[(\pi)(\sigma_y)(\sigma_z) + CA]}$$

\bar{U} = average wind speed (meters/second)

A = minimum cross-sectional area of the building (1600 square meters)

* Reference 2, Appendix C to the SER.

$$c = 0.5$$

$$p1 = 3.14$$

We have informed the applicant of our requirements regarding the methods of determining sigma-y, sigma-z and X/Q as described above. We have requested that the applicant include these methods in future amendments to the PSAR. We will include them in the technical specifications. We consider this matter to be resolved.

2.3.6 Conclusions

In the SER, we stated that the applicant must provide more detailed information on the program for control room monitoring of meteorological parameters, and that at least one additional year of onsite meteorological data must be submitted. Resolution of the control room monitoring program has been discussed in Section 2.3.3 of this report.

In Amendment 24, the applicant submitted additional joint frequency distributions of wind speed and direction at the 25-ft level by atmospheric stability (as defined by the vertical temperature gradient between the 25-ft and 250-ft levels) for the period May 1973 through April 1974.

These joint frequency distributions were submitted in accordance with the guidelines of Regulatory Guide 1.23 and are acceptable. As described in Sections 2.3.3, 2.3.4 and 2.3.5 of the SER, the applicant had utilized the meteorological assumptions of Regulatory Guide 1.4 and we had concluded, based upon our review of the meteorological data which had been submitted, that these

assumptions are adequately conservative. We have also reviewed the additional data submitted in Amendment 24.

Our evaluation, and that of our NOAA consultant (presented in Appendix B), confirm that the relative concentration values for short-term diffusion estimates based on the meteorological assumptions in Regulatory Guide 1.4 are adequately conservative. We consider this matter to be resolved.

4.0 REACTOR4.2 Mechanical Design4.2.1 Fuel

In the SER, we stated that the Single Rod Burst Tests had been completed and would be documented. The information with regard to the Single Rod Burst Tests has been documented in Westinghouse Topical Report WCAP-8289 (Proprietary) and WCAP-8290 (Non-Proprietary), "17 x 17 Design Fuel Rod Behavior During Simulated Loss of Coolant Accident Conditions," November 1974. Our review of this report has been completed and we concur with the conclusion made by Westinghouse that the test program indicates that the scaling down of rod geometries from the 15 x 15 design has no effect of practical significance on burst ductility and burst temperatures. We consider this matter to be resolved.

In the SER, we stated that the details of the fuel surveillance program would be reported in a supplement to the SER. Our evaluation of the 17 x 17 fuel design, which has been completed, included assessment of the engineering analysis, operating experience on similar fuel, confirmatory test results, technical specification requirements and a surveillance program to monitor the performance of the irradiated fuel. The surveillance program is essentially the last in a series of fuel design confirmations. The routine Westinghouse surveillance program consists of three monitorings; the power

distribution will be monitored using excore flux and incore moveable detectors, the coolant activity will be monitored for indications of loss of cladding integrity and, finally, the assemblies will be monitored by observations during the refueling operation.

Because the 17 x 17 fuel design is new and will be introduced into a number of plants in a short time, a comprehensive and generic surveillance program has been developed. The surveillance program is outlined in the Diablo Canyon FSAR. One feature of the program is the insertion of four lead burnup fuel assemblies into the two Surry reactors. One assembly in each of the Surry reactors will have removable fuel rods. These assemblies will have been dimensionally characterized prior to insertion and will be examined at subsequent refuelings. The fuel assemblies will be examined for dimensions, fretting, bowing, gamma activity, cladding integrity and surface deposits. The first of these assemblies should be examined near the end of 1975. This is compatible with the anticipated startup of Diablo Canyon Unit 1.

For four of the first reactors to employ a full core 17 x 17 fuel assembly, the surveillance program includes insertion of fuel assemblies with removable rods. This is shown in Table 4.1. These assemblies differ from the 17 x 17 assemblies in the Surry reactors only by the number of spacer grids. The Surry assemblies have seven

grids while the 17 x 17 standard design assemblies have eight grids. The slightly larger axial interval will cause slightly larger fuel rod deflections on the Surry assemblies than would the standard design 17 x 17 assemblies. The fuel rod damage mechanisms of interest (e.g., fretting from flow induced vibration) are service life dependent. Thus, the progression of any fuel damage can be acceptably monitored by the Westinghouse surveillance program.

A commitment to insert a removable rod assembly had been requested for the four reactors (Table 4.1) which were originally expected to be the first to operate with 17 x 17 fuel. Although four reactors are committed to the insertion of a pre-characterized assembly, only two of these are required to be examined as part of the generic 17 x 17 fuel surveillance program (Trojan and Diablo Canyon).

In Amendment 27, the applicant provided a commitment to perform visual inspections of the peripheral rods of those fuel assemblies in Diablo Canyon Unit 1 which have been permanently removed to the spent fuel storage pool for the purpose of discharge. This program would last until the entire initial load of fuel had been examined or a similar program has been completed on two other reactors in the United States and the NRC staff has approved cancellation of the program for Diablo Canyon Unit 1. If warranted

by the results of these visual examinations, further investigation of the pre-characterized removable rods would be conducted.

We have concluded that the fuel surveillance program for Diablo Canyon, as well as the generic fuel surveillance program for Westinghouse 17 x 17 fuel, is acceptable. We consider this matter to be resolved.

In the SER, 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. This information has not yet been submitted and this matter is not resolved. We will review this information when it is documented and will report the results of our review in a future supplement to the SER.

Subject to a favorable finding on the justification for applying the seven grid test results to eight grid fuel assemblies, we have concluded that the mechanical design of the Diablo Canyon fuel is acceptable.

4.4

Thermal and Hydraulic Design

In the SER, we stated that Westinghouse had submitted two topical reports: WCAP-8346, "An Evaluation of Fuel Rod Bowing," May 1974; and WCAP-8176 (Proprietary) and WCAP-8323 (Non-Proprietary), "Effect of Bowed Rod on DNB," May 1974, that describe the analytical techniques used to predict bowing and the method used for assessing the effect of bowing on thermal performance of the fuel. We stated that we were reviewing these topical reports.

Our review of WCAP-8176 and WCAP-8323 on the effect of a bowed rod on DNB has been completed. We have informed Westinghouse that these reports are acceptable. They describe the correlation between experimental results and calculational results obtained using the Westinghouse design methods. While we have concluded that the report provides an acceptable data base model for determining the effects of rod bowing on DNB heat flux during the first fuel cycle, additional information is required to demonstrate that the model adequately predicts the effect of rod bowing on DNB over subsequent fuel cycles. The applicant is aware of our need for additional information on this subject and is expected to provide it for our review in the near future. We consider the effects of rod bowing on DNB heat flux after the first fuel cycle to be unresolved. We will report the resolution of this matter in a future supplement to the SER prior to a decision concerning issuance of operating licenses for Diablo Canyon Units 1 and 2.

We have also completed our review of WCAP-8346, and have concluded that the fuel rod bowing predicted by the calculation model is acceptable for the 15 x 15 seven grid (rods-on-bottom) design based on observed bowing in irradiated fuel. We have also concluded that application of the same calculational methods is acceptable for design evaluation of fuel rod bowing in the 17 x 17 fuel assembly design. However, a 17 x 17 fuel assembly surveillance program is

needed to confirm the validity of this model. The program, discussed in Section 4.2.1 of this supplement, includes bowing measurements on irradiated fuel in the two Surry reactors and visual observations in the Diablo Canyon and Trojan reactors. The fuel rod bowing model will be reviewed as the data become available. If changes to the model are needed, the fuel rod bowing effects in Diablo Canyon will be reevaluated and we will make changes to the technical specifications if appropriate. We have concluded that the Diablo Canyon fuel design takes fuel rod bowing into account in an acceptable manner and consider that our concerns regarding this aspect of the Diablo Canyon design, as described in Section 4.4 of the SER, are resolved.

In the SER, we stated that, if the results of the non-uniform departure from nucleate boiling (DNB) 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 (DNBR) be increased 5 percent above that required to satisfy the 95/95 criterion. These results have not yet been documented. Our position on this matter remains unchanged.

In the SER, we stated that we would review the results of certain elements of the verification test program for the THINC code when they became available. In the event that sufficient

verification could not be obtained from the combined test and analytical programs, we stated that restrictions would be included in the technical specifications for Diablo Canyon to maintain required margins to fuel rod damage.

These results have 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. We have not yet completed our evaluation of these reports. We consider this matter to be unresolved. The resolution of this matter will be reported in a future supplement to the SER.

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.

TABLE 4.1

IRRADIATION SCHEDULE OF PRECHARACTERIZED FUEL
ASSEMBLIES FOR THE GENERIC 17 x 17 SURVEILLANCE PROGRAM

<u>Reactor</u>	<u>Fuel Cycle</u>	<u>Current^(a) Estimate of Startup</u>	<u>Number of Precharacterized 17 x 17 Assemblies</u>
Surry 1	2	Dec '74	2 ^(b)
Surry 2	2	May '75	2 ^(b)
Trojan ^{(c)(d)}	1	Jun '75	1
Diablo Canyon ^(d)	1	Jan '76	1
Farley	1	May '76	1
Sequoyah	1	Dec '77	1
Beaver Valley	1	Dec '75	✓
Salem	1	Nov '76	0
TOTAL			8

(a) As of December 19, 1974

(b) 7-grid assemblies, 1-each removable rod assembly

(c) Selected as lead plant prior to August 1974

(d) Currently expected to execute surveillance program

6.0 ENGINEERED SAFETY FEATURES
6.2 Containment Systems
6.2.1 Containment Functional Design

In the SER, we stated that the applicant had performed the containment subcompartment analyses using the Transient Mass Distribution (TMD) code with the augmented critical flow correlation, and that we had requested that the analyses be re-done using the more conservative TMD code without the augmented critical flow correlation. The applicant had provided us with an analysis of the pressure response within the pressurizer enclosures and the loop compartments using the non-augmented critical flow correlations and we were reviewing these analyses. We expected to receive similar analyses for the reactor coolant pipe annulus, reactor vessel annulus and lower reactor cavity.

We have completed our review of the analyses of the pressurizer enclosures and loop compartments using the TMD code without the augmented critical flow correlation. We have reviewed the noding arrangements used and the assumptions made and have concluded that the calculated maximum differential pressures for these compartments are acceptable. We consider this matter to be resolved.

In Amendments 25 and 26 to the FSAR the applicant has provided information showing the modeling assumptions and dimensions used for the analysis of the reactor cavity, reactor vessel annulus, and the reactor shield structure. The analysis of the reactor cavity and vessel annulus was performed using the TMD code without the augmented critical flow correlation. The analysis assumed a limited displacement rupture of a reactor coolant system hot leg at the reactor vessel nozzle weld as the design basis accident. We have reviewed the noding arrangement used and the assumptions made concerning piping and vessel insulation behavior and conclude that the differential pressures calculated for the reactor cavity structures are acceptable for the assumed break. We will require, however, that the applicant 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.

With regard to the pressure response of the reactor coolant system pipe penetrations through the reactor shield structure, the applicant has not performed an acceptable analysis. We will require that the applicant (1) analyze the response of the piping penetration to a pipe break within the penetration using

either the TMD code without the augmented critical flow correlation or another acceptable method of analysis; or (2) justify that a reactor coolant system pipe break need not be postulated in the reactor shield structure pipe penetration according to the recommendations of Regulatory Guide 1.46.

We will report the resolution of these items in a future supplement to the SER.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Design Bases

In the SER, we stated that we had identified certain locations where a single incorrectly positioned motor-operated valve could result in total loss of the intended ECCS safety function. We also stated that the applicant would need to either lock out power to these motor-operated valves or modify the design to obtain an equivalent degree of protection.

In Amendment 27, the applicant provided a commitment to lock out power to those motor-operated valves identified by the NRC staff as affecting the function of the ECCS if spurious operation were to occur. We have reviewed this commitment and concluded that it provides an acceptable method of meeting the single failure criterion. The valves for which we will require locking out of power are those identified in item 7.12 of a request for additional information (see items 42 and 45 in Appendix A to the SER). The

appropriate requirements for removal and restoration of power will be included in the technical specifications.

We consider this matter to be resolved.

6.3.3 Performance Evaluation

In the SER, we stated that the requirements of 10 CFR Part 50.46(b), "Acceptance Criteria for ECCS for Light Water Cooled Nuclear Power Reactors," are applicable to Diablo Canyon Units 1 and 2, and that the applicant had submitted an analysis of the performance of the emergency core cooling system in accordance with this provision in Amendment 15 to the FSAR, dated August 2, 1974.

We have completed our review of the Westinghouse model, and on the basis of that review, the applicant will revise the analysis submitted in Amendment 15, and will submit a revised analysis for Diablo Canyon Units 1 and 2. We will review that analysis and will report the results in a future supplement to the SER prior to a decision concerning the issuance of operating licenses for Diablo Canyon Units 1 and 2.

6.3.5 Conclusion

In the SER, we stated that the acceptability of the ECCS was still being evaluated. Specifically, we stated that (1) the applicant would need to either lock-out power to certain motor-operated valves or modify the design to meet the single

failure criterion in another manner; and (2) the applicant's ECCS evaluation model and analysis results would need to be found acceptable. We consider the first item to be resolved as discussed above. We will report our conclusions regarding acceptability of the ECCS in a future supplement to the SER as discussed in Section 6.3.3 of this report.

- 7.0 INSTRUMENTATION AND CONTROLS
- 7.2 Reactor Trip System
- 7.2.2 Reactor Trip System Actuation Logic
- 7.2.2.1 Physical Separation

In the SER, we stated that we had found that the physical separation in the solid state protection system racks was inadequate. The input and output wire bundles terminated at a common connector of the isolation board. The center pins of the connector are not used and this provides separation at that point; however, there were no physical barriers or protection to separate the input and output wire bundles at the locations where they are in close proximity as they are routed from the connectors. The applicant agreed to install barriers to provide this separation.

The applicant has provided the physical barriers. The barriers separate the input and output wire bundles at the locations where they are in close proximity. We have reviewed this design change and reviewed the detailed drawings of the barriers during a site visit. We have concluded that the barriers separating the input and output wiring provide an adequate means of meeting the requirements of General Design Criteria 22 and 24 and the design is therefore acceptable. We consider this matter to be resolved.

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7.2.2.2 Electrical Isolation

In the SER, we stated that the photodiode isolators used to electrically isolate the safety signals from non-safety functions, as implemented in the solid state protection system, had not been qualified as acceptable isolation devices. We informed the applicant that tests would be required to verify that these isolators will meet the design basis requirements for system isolation. The applicant indicated that he would provide qualification test procedures and results.

The applicant has provided test procedures and results which verify that the photodiode isolators will meet their design basis requirements for system isolation. We have reviewed the information and concluded that the qualification tests of the photodiode isolators provide an acceptable basis for meeting the requirements of General Design Criterion 24 and the design is therefore, acceptable. We consider this matter to be resolved.

7.2.2.4 Conclusions

In the SER, we stated that the solid state protection system would be acceptable, providing that: (1) the photodiode isolators were adequately qualified; (2) adequate separation or barriers were provided for the input and output wiring; and (3) the seismic qualification program conformed to our requirements. We consider the first two items to be resolved, as discussed above. We consider the third item to be unresolved.

We will report the final resolution of the seismic qualification program in a supplement to the SER.

7.3

Engineered Safety Features Actuation System

7.3.4

Changeover from Injection to Recirculation Mode

In the SER, we stated that the design would be acceptable, subject to satisfactory resolution of (1) our concerns about lockout of power to ECCS motor-operated valves, and (2) compliance of the level instrumentation to IEEE Std 279-1971. The first item has been resolved as discussed in Section 6.3.1 of this supplement. Three water level instrument channels are provided for each refueling water storage tank. Two out of three logic is provided for regenerative heat removal pump trip and initiation of low level alarm. We have reviewed the implementation of the revised design in accordance with IEEE 279-1971, and concluded that it provides an acceptable means of meeting the requirements of General Design Criteria 20, 21, 22 and 23 and the design is, therefore, acceptable. We also consider this matter to be resolved.

Accordingly, as discussed in the SER, we have concluded that the operator will have sufficient time to perform the actions required to change over to the recirculation mode of operation, and the design meets the Commission's requirements, and is acceptable.

9.0 AUXILIARY SYSTEMS9.2 Fuel Storage and Handling9.2.3 Fuel Handling System

In the SER, we stated that we had not reached agreement with the applicant regarding storage of spent fuel in locations where it could not be damaged by a dropped fuel cask.

In Amendment 22, the applicant described a fuel storage scheme where only spent fuel assemblies which have decayed for 1000 hours or more would be stored where they could be damaged by a dropped fuel cask. The applicant stated that no more than 20 of these assemblies, all of which have decayed for more than 1000 hours, could be damaged by a dropped fuel cask.

We have reviewed this information and concluded that the following provisions, in addition to those proposed by the applicant, are necessary in order to determine that no more than 20 elements, all of which have decayed for more than 1000 hours, could be damaged by a dropped fuel cask:

- (1) No cask handling operation near the spent fuel pool will be performed unless spent fuel at any location in rack 5 or rack 6 has decayed for at least 1000 hours since shutdown. (The applicant proposed a similar restriction but it included all of rack 5 and part of rack 6 instead of all of both racks).

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- (2) Redundant interlocks will be provided on the cask handling crane to limit the position of the cask to the outermost corner of the cask recess area whenever the cask is being raised or lowered. (This is in addition to the restrictions already described in the FSAR.)

With these provisions, in addition to those already described in the FSAR, we concur that, based upon the configuration of the spent fuel pool, no more than 20 spent fuel assemblies, all of which have decayed for at least 1000 hours, could be damaged by a dropped fuel cask. The applicant has agreed to these additional provisions. The radiological consequences of a dropped fuel cask accident involving damage to 20 spent fuel assemblies which have decayed 1000 hours are acceptable, as discussed in Section 15.2.2 of this supplement. Accordingly, subject to the applicant's documentation of a commitment to the provisions described above, we consider this matter to be resolved.

15.0 ACCIDENT ANALYSES15.1 General

In the SER, we stated that we would require heaters for the auxiliary building charcoal filters for humidity control in the event of residual heat removal system leakage following a loss of coolant accident.

In Amendment 22 to the FSAR, the applicant agreed to provide such heaters, and described an analysis of the humidity of the air which enters the auxiliary building charcoal filters. In this analysis the applicant assumed that air entering the containment building would be heated 15°F before it reaches the auxiliary building charcoal filters. We have requested the applicant to provide justification for this assumption. We consider this matter unresolved. We will report the resolution of this matter in a future supplement to the SER.

15.2 Design Basis Accident Assumptions15.2.2 Fuel Handling Accident

Since the SER was issued, the applicant has submitted, in Amendment 27 of the FSAR, an analysis of the consequences of a fuel cask drop accident. We have evaluated the radiological consequences of this accident based on damage to a maximum of 20

spent fuel assemblies, all of which have decayed for at least 1000 hours (see Section 9.2.3 of this supplement). The assumptions used for this evaluation are given in Table 15-2, while the dose consequences are shown in Table 15-1. The calculated thyroid dose at the exclusion zone boundary is 18 Rem. The conclusion that no freshly irradiated fuel could be damaged is an important part of this evaluation since damaging a single assembly which has decayed for only 100 hours since shutdown could result in a thyroid dose of as much as 23 Rem at the exclusion zone boundary.

We have concluded that the dose consequences of this accident are well within the guideline values of 10 CFR Part 100 and are therefore acceptable.

TABLE 15-1

POTENTIAL OFFSITE DOSES DUE TO FUEL CASK DROP ACCIDENT

<u>Accident</u>	<u>Two Hour Exclusion Boundary (800 Meters)</u>		<u>Course of Accident Low Population Zone (9600 Meters)</u>	
	<u>Thyroid (Rem)</u>	<u>Whole Body (Rem)</u>	<u>Thyroid (Rem)</u>	<u>Whole Body (Rem)</u>
Fuel Cask Drop	18	<1	<1	<1

TABLE 15-2

FUEL CASK DROP ACCIDENT CALCULATION INPUT PARAMETERS

Shutdown Time	1000 hours
Number of Fuel Assemblies Damaged by Cask Drop	20
Power Peaking Factor	1.65
Iodine Fractions Released from Pool	
Elemental	75%
Organic	25%
Effective Filter Efficiency	
Elemental	90%
Organic	70%
<u>X/Q Values, sec/m³</u>	
0-2 hours @ 300 meters	5.45×10^{-4}
0-2 hours @ 9600 meters	2.2×10^{-5}
Watertight integrity of spent fuel pool maintained.	

22.0

CONCLUSIONS

In Section 22 of Supplement No. 1 to the SER, 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 have been resolved in this supplement. A revised status report on all of these items is given below:

- (1) The matters regarding meteorology have been resolved (Sections 2.3.3 and 2.3.6 of this report).
- (2) The applicant has provided additional information on the effects of tsunami waves 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 No. 1).
- (3) 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 (Sections 2.5.1 and 2.5.2 of Supplement No. 1).
- (4) The applicant has provided additional information on the potential consequences of pipe breaks outside containment; however, our evaluation of this information has not been completed (SER Section 3.6).
- (5) The applicant has not yet submitted required information confirming the seismic qualification of Category I instrumentation and electrical equipment (SER Sections 3.10 and 7.8).

- (6) Documentation has not yet been provided justifying the use of the results of tests of 7-grid assemblies to prove the acceptability of the 8-grid design (Section 4.2.1 of this report).
- (7) Our evaluation of the results of the single rod burst tests has been completed, and this matter is resolved (Section 4.2.1 of this report).
- (8) Our evaluation of the 17 x 17 fuel rod surveillance program has been completed, and this matter is resolved (Section 4.2.1 of this report).
- (9) The matters regarding uncertainties in the thermal and hydraulic design have been partially resolved (Section 4.4 of this report).
- (10) The matters regarding subcompartment pressure calculations using the Transient Mass Distribution (TMD) Program have been partially resolved (Section 6.2.1 of this report).
- (11) The item regarding lockout of power to certain motor-operated ECCS valves has been resolved (Sections 6.3.1, 6.3.5 and 7.3.4 of this report).
- (12) The applicant has not submitted a revised ECCS analysis in accordance with the Final Acceptance Criteria (FAC) (Sections 6.3.3 and 6.3.5 of this report).
- (13) The matters regarding physical and electrical separation in the solid state protection system have been resolved (Sections 7.2.2.1, 7.2.2.2 and 7.2.2.4 of this report).

- (14) The applicant has provided additional information regarding physical separation in the process analog system; however, our evaluation of this information has not been completed (SER Section 7.2.3).
- (15) Our evaluation of the Westinghouse generic ATWS model is not yet completed (SER Section 7.2.5).
- (16) The applicant has not provided adequate information to confirm the environmental qualification of Category I instrumentation and electrical equipment (SER Section 7.8).
- (17) The matters regarding the consequences of a postulated cask drop have been resolved (Sections 9.2.3 and 15.2.1 of this report).
- (18) Our evaluation of the proposed design modifications to the auxiliary building to bring about a reduction of the doses in the event of RHR leakage during the recirculation phase following a postulated LOCA has not been completed (Section 15.1 of this report).
- (19) The applicant has submitted information regarding the guidance in certain WASH documents which pertain to the operational quality assurance program; however, our evaluation of this matter has not been completed (Item 99 of Appendix A to Supplement No. 1 and Items 109 and 116 of Appendix A to this report).

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

APPENDIX ACONTINUATION OF THE CHRONOLOGY OF THE RADIOLOGICAL REVIEW

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|------|----------------------|---|
| 109. | January 30, 1975 | Submittal of Amendment No. 25 consisting primarily of additional information on subcompartment pressure calculations and the operational quality assurance program. |
| 110. | January 31, 1975 | Supplement No. 1 to the Safety Evaluation Report issued. |
| 111. | February 6, 1975 | Meeting with applicant to discuss seismic and environmental qualification of electrical equipment, and physical and electrical separation in the solid state protection and process analog systems. |
| 112. | February 7, 1975 | Meeting with applicant to discuss the geology and seismology of the Diablo Canyon site. |
| 113. | February 12, 1975 | Request No. 11 to applicant for additional information on geology and seismology. |
| 114. | February 18-19, 1975 | ACRS Subcommittee Meeting in San Luis Obispo, California. |
| 115. | March 3, 1975 | Meeting with PG&E management to discuss the geology and seismology of the Diablo Canyon site. |
| 116. | March 26, 1975 | Submittal of Amendment No. 26 consisting primarily of additional information on subcompartment pressure calculations and on recent guidance for quality assurance programs. |
| 117. | April 3, 1975 | Letter to applicant requesting additional information on the Emergency Core Cooling System. |

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118. April 4, 1975 Meeting with applicant to discuss items relative to the seismic design of Diablo Canyon.
119. April 7, 1975 Letter from applicant providing supplementary pages to report entitled "Westinghouse Protection System Noise Tests" which is referenced in Section 7.2 of the 23AR.
120. April 7, 1975 Letter from applicant regarding participation in Westinghouse program to evaluate corrosion resistance of possible alternate steam generator tube materials in an operating plant.
121. April 10, 1975 Second OL Prehearing Conference.
122. April 15, 1975 Letter from applicant transmitting 1974 Annual Financial Report.
123. April 23, 1975 Submittal of report concerning jet effects analysis for postulated pipe breaks outside containment.
124. April 28-May 2, 1975 Review of the Diablo Canyon seismic design at PG&E offices in San Francisco.
125. April 30, 1975 Submittal of Amendment No. 27 consisting of additional information required for the resolution of outstanding items in the SER.
126. May 1-2, 1975 Site visit and meeting related to electrical, instrumentation and control systems.



April 10, 1975

R32

Mr. Frank Schroeder, Acting Director
Division of Technical Review
United States Nuclear Regulatory
Commission
Washington, D.C. 20555

50-275/323

Dear Mr. Schroeder:

This refers to the letter of January 23, 1975 from William
P. Gammill, Chief, Site Analysis Branch, Division of Technical
Review requesting comments on the following:

Diablo Canyon 1 and 2
Pacific Gas and Electric Company
Final Safety Analysis Report
Amendment No. 24 dtd. 1/15/75

These comments are attached.

Sincerely,

Isaac Van der Hoven

Isaac Van der Hoven
Air Resources Laboratories

Attachment

cc: E.H. Markee, Site Analysis Br., USNRC

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eh

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5-2
Comments on

Diablo Canyon Site Units 1 and 2
Pacific Gas and Electric Company
Final Safety Analysis Report
Amendment 24 dated January 15, 1975

Prepared by

Air Resources Laboratories
National Oceanic and Atmospheric Administration

April 10, 1975

Using the latest available data of wind speed and direction (May 1973 through April 1974) at the 25-ft. level and corresponding temperature differences between 25 and 250 ft., we have calculated that for a ground level release there is a probability that a relative concentration of $1.8 \times 10^{-4} \text{ sec m}^{-3}$ will be exceeded 5% of the time at a distance of 800 m downwind. The atmospheric stability was categorized according to the temperature gradient criteria listed in AEC Safety Guide 1.23. The release time was assumed to be from 0-2 hours.

For the long-term annual release a ground source was assumed. This is in contrast to the 70-m release height assumed by the applicant (p. 2.3-24). Figure 2.2-22 shows the top of the tallest vent duct to be about 32 m above grade, while the adjacent reactor containment building is about 67 m in height above grade. We would thus assume that the vent release would be caught in the building wake. We calculate from the 25-ft. wind data that the highest annual average relative concentration will be at the 300-m site boundary towards the NW at a value of $2 \times 10^{-6} \text{ sec m}^{-3}$.

APPENDIX C

ERRATA TO THE SAFETY EVALUATION REPORT
AND SUPPLEMENT NO. 1

Safety Evaluation Report

<u>Page</u>	<u>Line</u>	
6-3	1	insert a comma after "system"
6-15	12	replace "aame" with "same"

Supplement No. 1

<u>Page</u>	<u>Line</u>	
22-2	5	replace "has been provided in a WCAP report" with "has not been provided"
22-2	8	replace "report" with "item"
22-3	20	replace "turbine" with "auxiliary"

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2 75/323
MAR 31 1975

William P. Ganmill, Chief, Site Analysis Branch, TR
THRU: J. C. Stepp, Leader, Geology & Seismology Section, SAB, TR

INFORMAL MEMO RE: TELEPHONE CONVERSATION WITH DOUGLAS HAMILTON OF
EARTH SCIENCE ASSOCIATES (CONSULTANTS TO PG&E RE: DIABLO CANYON POWER
PLANT).

DATE: 24 March 1975

To keep you informed of progress on the Diablo Canyon work, a summary of
my 1:00 pm call to Douglas Hamilton follows.

After our trip to Houston of 18 and 19 of March to observe Shell Oil Co.
and Western Geophysical Company data, D. Hamilton and C. R. Willingham
were to integrate recently released or soon-to-be released USGS offshore
seismic data with their previous work. They were to keep in mind the
Shell data we observed and the availability of Western Geophysical data.

Progress to date:

- (1) Track charts of the Western Geophysical data and of the Shell data
that was shown to us have been added to a master track chart of the
area. Thus duplication of coverage can be determined.
- (2) Hamilton is recommending to PG&E that all Western data south of the
latitude of Arroyo Grande be purchased and that line 12 opposite
the site be purchased with additional processing eg. restacking,
special filtering and migration. The latter will help establish
the precise location of the "Hosgri" fault offshore from the plant.
- (3) The U.S.G.S. 1972 Bartlett cruise data has been ordered and should
arrive in microfilm form tomorrow. Several days will be required
to enlarge it to a working paper size. This is the data that was
collected and analyzed under the direction of Eli Silver and Roland
Von Huene.
- (4) The U.S.G.S. 1972 or earlier Polaris Cruise Seismic data has been
requested. However, they have been informed that the data will not
be released for several weeks. This data was collected under the
direction of Holly Wagner and Steve Wolfe.

There are three unknowns in the time table of required events which will
effect the completion of a combined analysis of all data.

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OFFICE →	TR: SAB <i>RBN</i>	TR: SAB <i>CS</i>				
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MAR 31 1975

1. When will the Polaris data become available?
2. When can the Shell data be observed?
3. When will they receive the Western Geophysical data.

The last item is presently of the least concern.

Carl Stepp has proposed a meeting at Menlo Park with Hamilton, Yerkes McKuen, Von Huene and us to review progress to date. Hamilton believes such a meeting could be held but without the Polaris information.

Renner B. Hofmann
Seismologist
Site Analysis Branch
Division of Technical Review
Office of Nuclear Reactor Regulation

cc: H. Denton
F. Schroeder
D. Allison
O. Parr

OFFICE ➤						
SURNAME ➤						
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