JUL 14 1977

SUPPLEMENT NO. 6

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/10

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1.0 INTRODUCTION

The Commission's Safety Evaluation Report in the matter of Pacific Gas and Electric Company's application for operating licenses for 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 evaluation had not been completed. Supplement Nos. 1, 2, 3, 4, and 5 to the Safety Evaluation Report, issued on January 31, 1975, May 9, 1975, September 18, 1975, May 11, 1976, and September 10, 1976, respectively, documented the resolution of certain outstanding items, and summarized the status of the remaining outstanding items.

The purpose of this supplement is to further update the Safety Evaluation Report by providing our evaluation of certain matters that were not resolved when Supplement No. 5 was issued and our evaluation of new issues that have arisen since Supplement No. 5 was issued. Each of the following sections of this supplement is numbered the same as the corresponding section of the Safety Evaluation Report that is being updated. A summary of the remaining outstanding issues, which will be addressed in future supplements to the Safety Evaluation Report, is presented in Section 22.0 of this supplement.

Appendix A to 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 notes the status of items in the Advisory Committee on Reactor Safeguards Report on Generic Items in relation to Diablo Canyon, Units 1 and 2.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.6 Protection Against the Dynamic Effects Associated With the Postulated Rupture of Piping

We stated in the Safety Evaluation Report that we would provide our evaluation of postulated pipe rupture outside containment for Diablo Canyon, Units 1 & 2 in a future report.

In response to our requests, the applicant has provided additional information in Amendment No. 44 to the Final Safety Analysis Report and in a letter dated July 6, 1977. We have determined from our review of this information that the design of Unit No. 1 is acceptable. However, we have not completed our review of Unit 2. Our evaluation is provided below.

The Unit 1 design accommodates the effects of postulated pipe breaks and cracks in high energy fluid piping systems outside containment with respect to pipe whip, jet impingement and resulting reaction forces, and environmental effects. The means used to protect safety-related systems and components include physical separation, enclosure in suitably designed structures or compartments, physical pipe enclosures, pipe whip restraints, and equipment shields. The applicant analyzed high energy piping systems for the effects of pipe whip, jet impingement, and environmental effects on safety-related systems and structures. For moderate energy systems, protection from the jet and environmental effects due to critical cracks was incorporated into the design.

Unit 1 has the ability to sustain a high energy pipe failure coincident with a single active failure and retain the capability of safe cold shutdown. For postulated pipe failures, the resulting environmental effects will not preclude the habitability of the control room, the accessibility of other areas that have to be manned during and following an accident, or the loss of function of electric power supplies, controls and instrumentation needed to complete a safety action.

Based on our review, we find that the applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. The design of Unit No. 1 meets the criteria set forth in our letter dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," regarding the protection of safety-related systems and components from a postulated high energy line break, and the Branch Technical Position APCSB 3-1 regarding the protection of safety-related systems and components from a postulated moderate energy line failure.

Therefore, we have concluded that the Unit No. 1 design for the protection of safety-related equipment from the effects of postulated piping failures outside containment is acceptable.

There will be some design differences between Unit 1 and Unit 2 in the methods provided for protecting against postulated pipe breaks.

We have requested the applicant to provide us with additional information describing where Unit 2 will be different and, where there are differences, describing the protection provided in Unit 2. Our evaluation for Unit 2 will be presented in a future supplement to the Safety Evaluation Report.

4.0 REACTOR

4.4 Thermal and Hydraulic Design

Introduction

In Supplement No. 4 to the Safety Evaluation Report we indicated that we had reviewed the Diablo Canyon core design calculations with respect to margins from departure from nucleate boiling (DNB). Two areas of our evaluation were not completed. However, we had examined the potential penalties associated with the two open areas and compared the potential penalties with the margins available. On this basis we had established, as an interim position, acceptable technical specification limits to be employed pending completion of our evaluations.

Since Supplement No. 4 was issued, we have completed our evaluation in one of the two areas that was open. However, in another area, that was closed in Supplement No. 4, we have received additional information that may change our conclusion. Accordingly, we have considered the potential penalties associated with the areas that are now open and compared them with the margins that are available. We have determined, as an interim position, acceptable technical specification limits to be employed pending further experimental work.

Our evaluation is provided below.

Effect of Nonuniform Heating on DNB

In Section 4.4 of Supplements 2, 3, and 4 we stated that we were reviewing the results of DNB tests involving nonuniform heating. These results were reported in WCAP-8536 (Proprietary) and WCAP-8537 (Nonproprietary),

"Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometries with 22 Inch Grid Spacing." We also indicated that, unless our evaluation of this matter was completed before the final technical specifications for Diablo Canyon were ready, we would require that the minimum allowable departure from nucleate boiling ratio (DNBR) be increased by 5 percent above that required to satisfy the 95/95 criterion to account for the incomplete data base.

We have now completed our evaluation of these topical reports. Our approval was documented in a letter to Westinghouse dated February 11, 1977. Accordingly, the 5 percent penalty that was associated with this open item is no longer appropriate.

Effect of Bowed Rod on DNB

In Section 4.4 of Supplement No. 4 we stated that we had completed our review of WCAP-8176 (Proprietary) and WCAP-8323 (Nonproprietary), "Effect of Bowed Rod on DNB." We had found these reports to provide an acceptable data base for predicting the effects of rod bowing on DNB heat flux.

However, in a letter dated August 13, 1976, Westinghouse reported data that indicated these methods for predicting the effect of fuel rod bowing on DNB may not contain adequate margins when unheated rods, such as instrument tubes, are present. Further experimental verification of these data is still in progress. As an interim position, we will require appropriate limitations in the technical specifications pending completion of the experimental work and our evaluation of it.

Amount of Rod Bowing

We stated in ection 4.4 of Supplement No. 4 that we had not completed our evaluation of the rod bowing model presented in Westinghouse Topical Reports WCAP-8691 (Proprietary) and WCAP-8692 (Nonproprietary), "Fuel Rod Bowing," dated December 1975. However, based on our evaluation to date, using conservative calculations, we had estimated the penalties on DNBR relative to the 95/95 criterion that would be necessary to account for the data, including an allowance for uncertainty in extrapolation of data from 15 x 15 fuel assemblies to predict the amount of bowing in 17 x 17 fuel assemblies.

Since we have not yet completed our evaluation of these topical reports, our position remains unchanged.

Evaluation

As an interim measure, to account for the items discussed above, we will include a burnup dependent penalty factor to be applied to the reactor operating limits in Section 3.2.3 of the Technical Specifications to reflect the reduced DNB conditions caused by increasing fuel rod bowing. The enthalpy hot channel factor, a parameter which varies inversely with DNB, is used to account for this penalty. This penalty is derived by extrapolating available test data and provides a conservative safety margin which we consider acceptable for the Diablo Canyon plant.

Conclusion

We have concluded that, as an interim position, the Diablo Canyon cores can be operated safety utilizing appropriate technical specification limits discussed above to account for uncertainties with regard to rod bowing and DNB.

We consider this matter resolved.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary Components

5.2.2 Overpressurization Protection

Introduction

In the Safety Evaluation Report we concluded that the overpressure protection for the reactor coolant system, as provided by the three spring loaded pressurizer safety valves, was acceptable. (In addition, Diablo Canyon employs three power operated pressurizer relief valves. However, they were not relied upon and were not discussed in the Safety Evaluation Report.)

Since then, we have reviewed the possibility of an overpressure transient at temperatures below operating temperatures during startup or shutdown when the allowable pressure is lower than the pressurizer relief valve setpoints. The applicant has proposed interim measures to minimize the possibility of such overpressure transients below normal operating temperatures. We have reviewed the interim measures and found them acceptable for use on Unit 1 during the first fuel cycle. Our evaluation is provided below.

Background

Incidents known as pressure transients (events that have exceeded the technical specification temperature-pressure limits of the reactor vessel) were discussed in a technical report issued in November 1976, "NUREG-0138, "Staff Discussion of Fifteen Technical Issues listed in

Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff." The report concluded in part that pressure transients are of concern during startup and shutdown because, at these relatively low temperatures, the vessel has less toughness than at operating temperatures and irradiation increases the temperature at which steel attains maximum toughness. The majority of these incidents occurred during cold shutdown while the system was in a solid-water condition (no bubble in the pressurizer).

The allowable pressure limits at lower temperatures are determined in accordance with Appendix G to 10 CFR Part 50 and are included in the technical specifications for operating licenses. The limits change during the life of the plant as the pressure vessel becomes irradiated. Because it would be impractical to change these limits continuously, they are calculated for an extended period of time. Thus, the limits in effect at a given time may be based on properties expected in the vessel five or more years in the future, making them conservative during the early portion of this period. The report concluded that large safety margins exist for unirradiated reactor vessels and new plants can be permitted to be licensed under existing safety criteria. Nevertheless, we have concluded that administrative procedures and overpressure protection devices should be upgraded to reduce the likelihood of future overpressure transients.

Evaluation

The applicant has described several modifications to his administrative procedures, design, and operator training to minimize the likelihood of overpressure transients. These measures were documented in Amendment 48 to the Final Safety Analysis Report and in a letter dated July 5, 1977. They include the following:

(1) Operator Training - Operators have been briefed on the types of events that could cause overpressurization and changes have been made in the operating procedures to minimize the probability of such events.

(2) Residual Heat Removal Relief Valves - The residual heat removal system will not be isolated from the reactor coolant system when the plant is in a solid-water condition, making the residual heat removal system relief valves available for pressure relief. These valves are set to relieve at pressures lower than the Appendix G pressure limits for the Unit 1 pressure vessel.

(3) Steam Bubble - A steam bubble will be formed in the pressurizer when the reactor coolant temperature reaches 160 degrees Fahrenheit during plant heatup. During the cooldown the bubble will be collapsed when the reactor coolant temperature reaches 160 degrees Fahrenheit. This will minimize the amount of time in a water-solid condition.

(4) Operating Procedures - As many operations as possible will be performed while the plant is not in a solid-water condition.

In addition, instructions have been added to the operating procedures which only permit slow changes in the residual heat removal system flow rate.

(5) Letdown Line - The letdown flow control valve will be placed in manual control in a fully open position when the reactor coolant system is in a water-solid condition. Additionally, an orifice relief valve on this line is available as another pressure relieving mechanism.

(6) Reactor Coolant Pumps - At least one pump will be operating when the reactor coolant system temperature is above 160 degrees Fahrenheit. If all the pumps should be shut down for any reason, appropriate restrictions will be employed on restarting to avoid possible pressure transients from injecting any accumulation of cold pump seal injection water into a warm reactor coolant system.

(7) Accumulators - The accumulator isolation valves will be closed and power will be removed and locked out whenever the primary system is in a solid-water condition.

(8) Alarm - An interim alarm will be installed to annunciate on the main control board whenever the reactor coolant system pressure approaches within 50 pounds per square inch of the allowable pressure as calcuated by a function generator. The pressure calculated by the function generator conservatively bounds the Appendix G limits.

(9) Safety Injection Pumps - Safety Injection Pumps are kept inactive whenever the primary system is solid. Power is removed by physically racking out the pump motor circuit breakers. When the pump motors are de-energized, an alarm is actuated in the control room indicating the status.

We have reviewed these proposed administrative and design changes and find them acceptable as an interim measure to minimize the likelihood of an overpressure transient at low temperature.

In addition, we have reviewed the material properties of the Unit 1 reactor vessel. In light of the margins available during the first fuel cycle we have determined that credible overpressure transients would not cause reactor vessel failure.

Conclusions

Based on our evaluation as described above, we have concluded that the interim measures proposed by the applicant to minimize the possibility of an overpressure transient are acceptable for use on Unit 1 during the first fuel cycle.

The applicant is a member of a group of utilities that is developing a long-term solution for this issue. The design modifications under consideration would employ the power operated pressurizer relief valves, with lifting setpoints programmed as a function of reactor coolant temperature, to preclude violating the pressure limits of

Appendix G to 10 CFR Part 50. We will review the proposed long-term solution when the supporting analytical information is available. We will include a condition in any operating license for Unit 1 requiring that an acceptable long-term overpressure protection system be installed prior to the initiation of the second fuel cycle.

Since the Unit 2 reactor vessel has somewhat less resistance to brittle fracture at low temperature than the Unit 1 vessel we have not yet evaluated this matter for Unit 2. We will provide our evaluation in a future supplement to the Safety Evaluation Report.

5.2.8 Inservice Inspection Program

In the Safety Evaluation Report we stated that the inservice inspection program would comply with: (1) the ASME Boiler and Pressure Vessel Code, Section XI, including Addenda through Winter 1972, for Class 1 components and (2) Regulatory Guide 1.51 for inspection of Class 2 systems. Those provisions were acceptable at that time.

Since then, the Commission has revised its requirements for inservice inspection. The new requirements for ASME Class 1, 2, and 3 components are given in Section 50.55a(g) of 10 CFR Part 50, initially published February 12, 1976. We will require, in the technical specifications, that the applicant comply with this regulation.

In very general terms, the regulation requires that the inservice inspections conform, to the extent practical within the limitations of design, geometry and materials, to the requirements of a periodically updated edition of ASME Section XI.

In a letter dated May 23, 1977, we provided detailed guidance to the applicant on complying with the regulation. We also requested submittal of the applicant's proposed inservice inspection plan, along with justifications for any items where the applicant may determine that meeting ASME Section XI requirements would be impractical.

We requested the submittal by September 30, 1977, which should provide adequate time for our evaluation of any requests for relief from ASME Section XI requirements prior to the need to implement the inservice inspection program at the Diablo Canyon plant.

The technical specifications will require conformance to 10 CFR 50.55a(g) and, accordingly, we have concluded that this provides an acceptable basis for the inservice inspection and testing program to comply with Criterion 32 of the General Design Criteria.

We consider this matter resolved.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In Section 6.2.1 of the Safety Evaluation Report we presented our evaluation of the applicant's calculations of the containment presssure response to a postulated loss-of-coolant accident and a postulated main steam line break. (Subcompartment pressure responses were open items in the Safety Evaluation Report. Later, they were resolved in Section 6.2.1 of Supplements 2 and 3.)

Our recent evaluations of a postulated main steam line break inside containment have indicated a potential concern in two areas: (1) The peak calculated containment pressure and temperature, and (2) The environmental qualification of safety-related equipment located inside containment that must function.

In a letter to the applicant dated June 1, 1977 we requested additional information on these subjects. The applicant currently expects to respond to our request on about September 15, 1977. We will review the applicant's submittal and provide our evaluation in a future supplement to the Safety Evaluation Report.

6.3 Emergency Core Cooling System (ECCS)

We stated in Supplement No. 5 to the Safety Evaluation Report that the ECCS performance evaluation was acceptable, subject to satisfactory resolution

of two items concerning fuel rod bowing and the temperature of the reactor coolant in the upper reactor head. These matters are now resolved. Our evaluation of rod bowing and departure from nucleate boiling is presented in Section 4.4 of this supplement. Our evaluation of ECCS performance is presented below.

Emergency Core Cooling System Performance

In the analyses for Diablo Canyon, Units 1 and 2, an initial upper head temperature corresponding to the core inlet temperature had been used. These analyses had been accepted as described in Supplement No. 4 and Supplement No. 5.

In August 1976, Westinghouse reported that the fluid temperature in the upper head region may be higher. Recent data from operating facilities have indicated that the effective upper head temperature is between the cold leg and hot leg temperature. Westinghouse has performed sensitivity studies that show the calculated peak clad temperature increases for higher upper head temperatures. These were reported in a letter to the staff dated August 13, 1976. As a result, we have requested that the large breaks (which are most limiting) be conservatively reanalyzed with an upper head temperature corresponding to the hot leg temperature.

In Amendment 47 to the Final Safety Analysis Report the applicant submitted loss-of-coolant analyses for a large pipe break assuming the hot leg temperature would exist in the upper reactor head. The higher upper head temperature was used in a generic study provided in the Westinghouse Topical

Report, WCAP-8865, "Westinghouse ECCS-Four Loop Plant (17 x 17) Sensitivity Studies with Upper Head Fluid Temperature at T(hot)," dated October, 1976. This generic study indicated that the double-ended cold leg guillotine break was still the most limiting break for Westinghouse four-loop plants.

The applicant has referenced the loss-of-coolant analyses performed for Salem Nuclear Generating Station, Unit No. 1 (Docket No. 50-272), to establish that the worst break size for Diablo Canyon, Units 1 and 2, remains the double-ended guillotine break with a discharge coefficient of 0.6. The Salem Unit 1 analyses are for a design similar to Diablo Canyon, Unit 1. We have reviewed them and found them to be acceptable and, therefore, they are acceptable for our review of Diablo Canyon, Unit 1 and 2. These calculations satisfy the requirements of 10 CFR 50.46 to analyze a spectrum of different breaks to determine the worst case.

The applicant has established that the double-ended cold leg guillotine break loss-of-coolant accident analysis using a discharge coefficient of 0.6 presented for Unit No. 2 provides bounding results for both Unit 1 and Unit 2. The analyses were performed with the October, 1975 version of the Westinghouse Evaluation Model. This was documented in Westinghouse Topical Reports WCAP 8522 (Proprietary) and WCAP 8523 (Nonproprietary), "Westinghouse ECCS Evaluation Model, October 1975 Version," November 1975. We have reviewed and approved this model. Our approval was documented in a letter to Westinghouse dated May 13, 1976.

The analyses, performed with a total peaking factor of 2.32 at a power level of 102 percent of 3411 megawatts thermal, identified the worst break as a double-ended cold leg guillotine with a discharge coefficient of 0.6. The calculated peak clad temperature is 2130 degrees Farenheit, which is below the acceptable limit (2200 degrees Farenheit) specified in 10 CFR Part 50.46. In addition, the calculated maximum local metal/water reaction of 6.76 percent and a total core-wide metal/water reaction of less than 0.3 percent are well below the allowable limits of 17 percent and 1 percent, respectively, as delineated in 10 CFR 50.46.

The applicant's calculation of the containment pressure transient was based on the Westinghouse Evaluation Model and revised mass and energy release data. All plant dependent input parameters remained identical to those assumed for the emergency core cooling system evaluation previously performed and reported in Amendment 35 to the Final Safety Analysis Report. Since the reanalysis of the containment pressure transient was done using the Westinghouse Evaluation Model and the same plant dependent input parameters, we reaffirm our conclusions as stated in Supplement No. 4 stated that the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 and, therefore, are acceptable.

The single failure evaluation of the emergency core cooling system has been previously reviewed and found acceptable, as stated in Supplement No. 5.

The evaluations of long term boric acid buildup and submerged valves have been previously reviewed and found acceptable as stated in Supplement No. 4. A 5 percent spike penalty was discussed in Section 6.3 of Supplement No. 4 in connection with rod bow. This is no longer appropriate for the ECCS performance analysis in light of our current evaluation of rod bow, which is presented in Section 4.4. of this Supplement.

The applicant has stated that the reactor core and internals have been designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following a postulated loss-of-coolant accident. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling and the cladding oxidation limits of 17 percent are not exceeded during or after quenching.

Based on our review as described in the Safety Evaluation Report and Supplements Nos. 4 and 5 and in this supplement, we have concluded that (1) the LOCA analyses that were performed are wholly in accordance with the requirements of Appendix K to 10 CFR Part 50, (2) the ECCS cooling performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46, (3) ECCS cooling performance will be adequate despite any postulated failure of a single active component, (4) adequate systems are available to provide long-term core cooling to the reactor vessel and, therefore, (5) the emergency core cooling system design is acceptable.

7.0 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.5 Anticipated Transients Without Scram

In the Safety Evaluation Report and in Supplement No. 4 we stated that we had not completed our evaluation of the information which had been submitted by the applicant concerning anticipated transients without scram. The current status of this matter is described below.

Our position with respect to anticipated transients without scram is provided in the technical report, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, dated September 1973. Unit 1 was classified by the staff as a "I.C." facility as defined in WASH-1270; for this Unit, WASH-1270 indicates an analysis describing and evaluating the consequences of a postulated anticipated transient without scram would be acceptable. Unit 2 was classified as a "I.B." facility; for this Unit, WASH-1270 indicates a program to incorporate any design changes necessary to assure that the consequences of anticipated transients without scram would be accepable. The applicant submitted the information described by WASH-1270 in a letter dated October 1, 1974. In that letter the applicant referred to two Westinghouse topical reports, WCAP-8330, "Westinghouse ATWS Analysis," dated August 1974 and WCAP-7706, "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," dated July 1971. The applicant stated that these reports contain the necessary analyses

and that they are applicable to the Diablo Canyon Nuclear Power Plant. In addition, the applicant stated that Unit 1 as well as Unit 2 would be considered as a category "I.B." facility.

We evaluated these Westinghouse topical reports and presented our conclusions in a report, "Status Report on Anticipated Transients Without Scram for Westinghouse Reactors," dated December 9, 1975. This report described a number of outstanding issues. We have discussed the status report with the Advisory Committee on Reactor Safeguards. Since then Westinghouse has submitted additional information. We are evaluating this information and we plan to issue staff positions in September 1977.

If our review shows that design modifications are necessary to mitigate the consequences of anticipated transients without scram we will require that they be implemented at Diablo Canyon, Units 1 and 2.

The effect of this matter on the review and licensing process at this time does not change the conclusion as stated in WASH-1270 that limitations on operation on this account are not necessary or appropriate. This conclusion is based on our determination that the likelihood of an anticipated transient without scram event is very low considering the number of plants now in operational status, or expected to come into operation before our requirements can be fully implemented.

8.0 ELECTRIC POWER

8.1 General

Effects of Degradation of Offsite Power Voltage

We were informed on July 20, 1976 by the Northeast Nuclear Energy Company that following a trip on July 5, 1976 at the Millstone Nuclear Station, Unit 2 (Docket No. 50-336), equipment failures occurred during a degraded grid voltage condition.

On July 27, 1976, we notified all licensees with operating reactor facilities of the events which had occurred at the Millstone, Unit No. 2 facility, and requested the licensees to investigate the potential for equipment failures for degraded voltage conditions.

As a result of our initial investigation and evaluation of the events occurring at Millstone, Unit 2, we considered it necessary to require all plants presently in review for an operating license to conduct a thorough evaluation of the problem and to submit formal reports.

In a letter dated June 6, 1977, we requested that the applicant evaluate the Diablo Canyon, Units 1 and 2 design for the Class IE electrical distribution system to determine whether the operability of safety related equipment, including associated control circuitry and instrumentation, can be adversely affected by short term or long term degradation in the offsite power system.

The applicant currently expects to submit a response by about October 15, 1977. We will evaluate the applicant's response and will report the resolution of this matter in a future supplement to the Safety Evaluation Report.

Subject to resolution of this matter, we reaffirm our conclusions as stated in the Safety Evaluation Report that the electrical power system for Diablo Canyon, Units 1 and 2 is acceptable.

9.0 AUXILIARY SYSTEMS

9.5 Air Conditioning, Heating and Ventilation Systems

9.5.5 Diesel Generator Compartments

In the Safety Evaluation Report we presented our evaluation of the diesel generator compartment ventilation system.

In addition to the ventilation features described in the Safety Evaluation Report, the diesel generators draw some cooling air from the main turbine bay section of the turbine building. This air is essential for cooling the generators.

The applicant has informally committed to install a normally closed fire door to isolate the diesel generator compartments from the main section of the turbine building. An alternate source of cooling air will be provided, via ductwork, from the area above the diesel generator compartments. Since this modification will eliminate the possibility of a fire in the main section of the turbine building causing overheating of the diesel generators, we find the applicant's commitment acceptable.

We will review the details of this modification when they are submitted and provide our evaluation in a future supplement to the Safety Evaluation Report.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

Background - Appendix I to 10 CFR Part 50

In the Safety Evaluation Report we presented our evaluation of the radioactive waste management systems. We found them to be acceptable and capable of maintaining the amounts of radioactivity in effluent streams as low as practicable.

In Supplement No. 4 to the Safety Evaluation Report we indicated that the Commission had adopted a new regulation, Appendix I to 10 CFR Part 50. Appendix I provided numerical guidelines as to what constituted maintaining the amount of radioactivity in effluent streams as low as reasonably achievable and required a reevaluation of the liquid and gaseous radioactive waste management systems.

The applicant submitted the information necessary for our evaluation in Amendment 44 to the Final Safety Analysis Report, dated July 29, 1976 and in a letter dated July 30, 1976. The applicant elected to demonstrate compliance with an option allowed by a September 4, 1975 amendment to Appendix I. This option allows certain applicants to demonstrate compliance with the Annex to Appendix I, "Concluding Statement of Position of the Regulatory Staff (Docket-RM-50-2)," in lieu of performing the detailed cost-benefit study that would otherwise be required by Section II.D of Appendix I.

We have completed our evaluation which is described below.

Evaluation - Appendix I to 10 CFR Part 50

We have evaluated the radioactive waste management systems proposed for Diablo Canyon, Units 1 and 2, to reduce the quantities of radioactive materials released to the environment in liquid and gaseous effluents. The radioactive waste management systems were previously described in Section 3.4 of the Final Environmental Statement (FES) dated May 1973, and in Section 11 of our Safety Evaluation Report dated October 16, 1974. Based on more recent operating data applicable to the Diablo Canyon Station, and on changes in our calculational model, we have generated new liquid and gaseous source terms to determine conformance with Appendix I. These values are different from those given in Tables 3.6 and 3.7 of the Final Environmental Statement.

The new source terms, shown in Tables 11.1 and 11.2, were calculated using the models and methodology described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1976. These source terms were used to calculate the doses described below. Our evaluation of the dispersion of radionuclides in and the disposition of radionuclides from the atmosphere were based on analyses using the methodology provided in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," March 1976. The mathematical models used to perform the dose calculations are contained in Regulatory

Guide 1.109, "Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Implementing Appendix I," March 1976.

Included in our analysis are dose evaluations of three effluent categories: (1) pathways associated with liquid effluent releases to the Pacific Ocean, (2) noble gases released to the atmosphere, and (3) pathways associated with radioiodines, particulates, carbon-14 and tritium released to the atmosphere.

The dose evaluation of pathways associated with liquid effluents was based on the maximum exposed individual. The dietary and living habits for an adult individual included the consumption of 21 kilograms per year of fish harvested in the immediate vicinity of the cooling water discharge into Diablo Cove, and recreational use of its shoreline for 10 hours per year. There are no drinking water sources receiving Diablo Canyon Station liquid effluents. The maximum dose commitment resulting from exposure to water from Diablo Cove was estimated to be 0.024 mrem per year (total body) and 0.077 mrem per year (thyroid) for an adult.

The dose evaluation of noble gases released to the atmosphere included a calculation of beta and gamma air doses at the site boundary and total body and skin doses at the residence having the highest dose. The maximum air doses at the site boundary were found at 0.5 mile north-northwest relative to Diablo Canyon, Unit No.s 1 and 2. The

location of maximum total body and skin doses were determined to be at a residence located 1.5 miles north-northwest of the station.

The dose evaluation of pathways associated with radioiodine, particulates, Carbon-14 and tritium released to the atmosphere was also based on the maximum exposed individual. This individual is a child whose diet included the consumption of 41 kilograms per year of beef from an animal grazing year-round located 0.5 mile north-northwest of Diablo Canyon, Unit Nos. 1 and 2.

As shown in Table 11.1, the expected quantity of radioactive materials released in liquid effluents from Diablo Canyon, Units 1 and 2, will be less than 5 Curies per year per reactor (0.34 Curies per year per reactor), excluding tritium and dissolved gases, in conformance with the amendment to Section II.D of Appendix I. The liquid effluents released from Diablo Canyon, Units 1 and 2, will not result in an annual dose or dose commitment to the total body or to any organ of an individual, in an unrestricted area from all pathways of exposure, in excess of 5 mrem as shown in Table 11.3.

Based on our evaluation of the gaseous radwaste management systems, the total quantity of radioactive materials released in gaseous effluents will not result in an annual gamma air dose in excess of 10 mrads and a beta air dose in excess of 20 mrads at every location which could be occupied by individuals near ground level, and at or beyond the site boundary as shown in Table 11.3. As shown in Table 11.2,

the annual total quantity of Iodine-131 released in gaseous effluents will be less than 1 Curie per reactor (0.076 Curie per year per reactor) in conformance with the September 4, 1975 amendment to Appendix I. The annual total quantity of radioiodine and radioactive particulates released in gaseous effluents from Diablo Canyon, Units 1 and 2, will not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem as shown in Table 11.3.

Based on our evaluation, the radwaste treatment systems proposed for Diablo Canyon, Units 1 and 2, are capable of maintaining releases of radioactive materials in liquid and gaseous effluents during normal operation at doses which will not exceed the design objectives of Sections II.A, B and C of Appendix I of 10 CFR Part 50.

Our evaluation also shows that the applicant's proposed design of Diablo Canyon, Units 1 and 2, satisfies the design objectives set forth in the option provided by the Commission's September 4, 1975 amendment to Appendix I and, therefore, satisfies Section II.D of Appendix I of 10 CFR Part 50.

We conclude that the liquid and gaseous radwaste treatment systems will reduce radioactive materials in effluents to "as low as is reasonably achievable levels" in accordance with 10 CFR Part 50.34a and, therefore, are acceptable.

Based on our evaluation as set forth in our Safety Evaluation Report and our evaluations as stated above in this Supplement, we find the liquid, gaseous and solid radwaste systems and associated process and effluent radiological monitoring and sampling systems for Diablo Canyon, Units 1 and 2, to be acceptable.

We consider this matter resolved.

TABLE 11.1

CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FROM DIABLO CANYON, UNITS 1 AND 2

CURIES PER YEAR PER REACTOR

Nuclide	Ci/yr/reactor
Corrosion & Activation Products	
Chromium-51	3.3(-4)
Manganese-54	1.1(-3)
Iron-55	5 (-4)
Iron-59	2.2(-4)
Cobalt-58	8 (-3)
Cobalt-60	9.3(-3)
Zirconium-95	1.4(-3)
Niobium-95	2 (-3)
Fission Products	
Bromine-83	1 (-4)
Rubidium-86	9 (-5)
Strontium-89	8 (-5)
Yttrium-91	2 (-5)
Zirconium-95	1 (-5)
Niobium-95	2 (-5)
Molybdenum-99	7.5(-4)
Technetium-99m	8.2(-4)
Ruthenium-103	1.5(-4)
Ruthenium-106	2.4(-3)
Silver-110m	4.4(-4)

Nuclide	Ci/yr/reactor
Fission Products (Continued)	
Tellurium-127m	8 (-5)
Tellurium-127	8 (-5)
Tellurium-129m	2.7(-5)
Tellurium-129	1.7(-4)
Iodine-130	4 (-5)
Iodine-131	7.5(-2)
Tellurium-132	2.8(-2)
Iodine-132	1.7(-3)
Iodine-133	1.1(-2)
Cesium-134	8.4(-2)
Iodine-135	1.9(-3)
Cesium-136	9.6(-3)
Cesium-137	7.6(-2)
Barium-137m	4.9(-2)
Barium-140	2 (-5)
Lanthanium-140	2 (-5)
Cerium-141	1 (-5)
Cerium-144	5.2(-3)
All Others	7 (-5)
Total (except Tritium)	3.4(-1)
Tritium	710

* Exponential notation: $1(-4) = 1 \times 10^{-4}$

** Nuclides whose release rates are less than 10 Ci/yr/reactor are not listed individually, but are included in the category "All Others"

TABLE 11.2

CALCULATED RELEASES OF RADIOACTIVE MATERIALS
IN GASEOUS EFFLUENTS FROM
DIABLO CANYON, UNITS 1 AND 2

CURIES PER YEAR PER REACTOR

Radionuclide	Reactor Building	Auxiliary Building	Turbine Building	Air Ejector	Decay Tanks	Total
Krypton-83m	a	a	a	a	a	a
Krypton-85m	2	2	a	1	a	5
Krypton-85	6	a	a	a	250	250
Krypton-87	a	1	a	a	a	1
Krypton-88	2	4	a	3	a	9
Krypton-89	a	a	a	a	a	a
Xenon-131m	10	a	a	a	22	32
Xenon-133m	20	2	a	1	a	23
Xenon-133	1900	110	a	68	170	2300
Xenon-135m	a	a	a	a	a	a
Xenon-135	10	6	a	4	a	20
Xenon-137	a	a	a	a	a	a
Xenon-138	a	a	a	a	a	a
Iodine-131	2.6(-3)	4.4(-2)	1(-3)	2.8(-2)	a	7.6(-2)
Iodine-133	2.8(-3)	6.2(-2)	1.2(-3)	3.9(-2)	a	1(-1)
Manganese-54	2.4(-2)	1.8(-4)	c	c	4.5(-5)	2.3(-4)
Iron-59	8.3(-7)	6(-5)	c	c	1.5(-5)	7.6(-5)
Cobalt-58	8.3(-6)	6(-4)	c	c	1.5(-4)	7.6(-4)
Cobalt-60	3.8(-6)	2.7(-4)	c	c	7(-5)	3.4(-4)
Strontium-89	1.9(-7)	1.3(-5)	c	c	3.3(-6)	1.6(-5)
Strontium-90	3.3(-8)	2.4(-6)	c	c	6(-7)	3(-6)
Cesium-134	2.4(-6)	1.8(-4)	c	c	4.5(-5)	2.3(-4)
Cesium-137	4.2(-6)	3(-4)	c	c	7.5(-5)	3.8(-4)
Tritium-3	710	c	c	c	c	710
Carbon-14	1	a	a	a	7	8
Argon-41	25	c	c	c	c	25

less than 1.0 curie per year per reactor for noble gases and carbon-14, less than 10 curie per year per reactor for iodine

exponential notation: $1.4(-2) = 1.4 \times 10^{-4}$

less than 1 percent of total for this nuclide

radionuclides not listed are released in quantities less than those specified in notes a and c from all sources

TABLE 11.3

COMPARISON OF DIABLO CANYON, UNITS 1 AND 2, WITH APPENDIX I TO 10 CFR PART 50, SECTIONS II.A, II.B AND II.C (MAY 5, 1975(a) AND SECTION II.D, ANNEX (SEPTEMBER 4, 1975)(b)

Criterion	Appendix I(a) Design Objectives(c)	Annex(b) Design Objectives	Calculated Doses Units 1 or 2
Liquid Effluents			
Dose to total body from all pathways	3 mrem/yr/unit	5 mrem/yr/site	0.012 mrem/yr/unit
Dose to any organ from all pathways	10 mrem/yr/unit	5 mrem/yr/site	0/038 mrem/yr/unit
Noble Gas Effluents(d)			
Gamma dose in air	10 mrad/yr/unit	10 mrad/yr/site	0.22 mrad/yr/unit
Beta dose in air	20 mrad/yr/unit	20 mrad/yr/site	0.51 mrad/yr/unit
Dose to total body of an individual	5 mrem/yr/unit	5 mrem/yr/site	0.016 mrem/yr/unit
Dose to skin of an individual	15 mrem/yr/unit	15 mrem/yr/site	0.043 mrem/yr/unit
Radioiodines and Other Radionuclides Released to the Atmosphere(e)			
Dose to any organ from all pathways	15 mrem/yr/unit	15 mrem/yr/site	1.04 mrem/yr/unit

(a) Federal Register, V. 50, p. 19442, May 5, 1975.
(b) Federal Register, V. 40, p. 40816, September 4, 1975.
(c) Design Objectives given on a site basis. Therefore, these design objectives apply to 2 units at the site.
(d) Limited to noble gases only.
(e) Carbon-14 and Tritium have been added to this category.

13.0 CONDUCT OF OPERATIONS

13.6 Industrial Security

In the Safety Evaluation Report we stated that the industrial security program was acceptable. Since then, the Commission has revised the requirements for industrial security programs at commercial nuclear power plants. The new requirements are given in Section 73.55 of 10 CFR Part 73, published in the Federal Register on February 24, 1977. As required by the regulation, the applicant submitted an amended physical security plan on May 25, 1977 as Revision 2 to the plan. Revisions 3, 4 and 5 were submitted on June 3, 1977, June 15, 1977 and June 29, 1977, respectively.

The regulation requires that a revised security plan, that complies with the requirements of 10 CFR 73.55, except for certain items of construction and installation of equipment that is not already in place, must be implemented by May 25, 1977 or on the date of receipt of an operating license, whichever is later. The applicant has committed to implementing these portions of the security plan prior to the date of fuel loading.

A security plan that complies with all the requirements of the new regulation, including construction and installation items, must be implemented by August 24, 1978 or on the date of receipt of an

operating license, whichever is later. These construction and installation items will be implemented by the applicant prior to August 24, 1978, to upgrade the physical security measures for the plant site.

This on-going upgrading of physical security is consistent with the graded implementation permitted by the regulation and is, therefore, acceptable.

We have evaluated the amended security plan and a security plan review team has visited the plant site as part of this overall evaluation. As a result of our evaluation, certain areas have been identified where additional information is required before the amended security plan can be found in conformance with the regulation. The applicant has made commitments to modify the amended security plan such that the level of protection will be consistent with the performance requirements of Section (a) of §73.55.

We will review the details of implementing these commitments and provide our evaluation in a future supplement to the Safety Evaluation Report.

15.0 ACCIDENT ANALYSES

15.1 General

We have presented our evaluation of the accident analyses for Diablo Canyon in the Safety Evaluation Report and in Supplements 2 and 3 to the Safety Evaluation Report. Since then, we have requested detailed evaluations of fuel handling accidents inside the containment building for all plants under review for operating licenses. Our letter of March 11, 1977 forwarded the request for Diablo Canyon. The applicant provided this evaluation in Amendment 49 to the Final Safety Analysis Report. We have completed our evaluation, which is presented below.

The applicant has described the plant systems used to mitigate the consequences of a fuel handling accident inside containment. Five 110,000 cubic foot per minute fan coolers are radially arranged around the core cavity, with the inlets located about 30 feet above the surface of the refueling cavity pool and about 20 feet from the edge of the pool. During refueling, one or more of these fan coolers will be operating. These five inlets join into a common annular ring. The ring is purged by the containment purge exhaust fan during refueling operations at a rate of 55,000 cubic feet per minute through two 48-inch diameter butterfly valves, in series, and released to the atmosphere through the plant vent. A radiation monitor is located at the plant vent downstream of the isolation valves. Upon detection of a high radiation signal, the butterfly purge valves are automatically closed, isolating the containment.

Due to the location of the radiation monitor downstream of the isolation valves, a portion of any radioactivity would be released to the atmosphere before the isolation valves can close.

The applicant has performed an analysis of a fuel handling accident inside containment. In this analysis, he assumed that any activity released to the containment above the refueling cavity pool would be well mixed in a volume about 40 feet high and extending around the pool cavity and transfer canal. Due to the location of these inlets relative to the refueling cavity pool and transfer canal, we concur with the applicant that substantial mixing of any released activity would occur in the containment prior to its release from the plant vent. There would be additional dilution of contaminated air since a recirculation system exhausts between 10 percent and 50 percent of the air flow and returns the remainder to the containment atmosphere. The applicant has conservatively omitted this effect in his analysis. This would further reduce the doses shown in Table 15.1.

The applicant's proposed technical specification limit on closure time for the purge isolation valves is 10 seconds. An additional 9.0 seconds of transit time and instrument response time would elapse between the time the leading edge of any activity passed the isolation valves and the time the valves would begin to close. Therefore, the valves would close, isolating the containment, within about 19 seconds after initially detecting activity leaving the plant vent.

We have evaluated the applicant's analysis and performed an independent assessment of a postulated fuel handling accident inside containment. We have assumed a valve closure time of 10 seconds and a containment isolation time of 19 seconds. Our assumptions are described below and the calculated doses are shown in Table 15.1. We have concluded that, with the present plant systems, the radiological consequences in the event of a fuel handling accident inside containment would be well within the guideline values of 10 CFR Part 100, and therefore, acceptably low.

15.2 Design Basis Accident Assumptions

15.2.2 Fuel Handling Accident

The assumptions used in our evaluation to calculate the offsite doses from a refueling accident inside of containment are:

(1) Rupture of all fuel rods in one assembly.

(2) All gap activity in rods, assumed to be 10 percent of the noble gases and 10 percent of the iodine.

(3) Peaking factor of 1.65.

(4) The accident occurs 100 hours after shutdown.

(5) 99 percent of the iodine is retained by the pool water.

(6) Mixing in 33,600 cubic feet of containment air (0.0126 of the total containment volume).

(7) Exhaust rate of 229 cubic feet per second of contaminated air to the environment, contained in 917 cubic feet per second of containment purge flow.

(8) Isolation valve closure time of 10 seconds.

(9) Containment isolation time of 19 seconds.

(10) 0-2hour relative concentration values (X/Q) determined from onsite meteorological program.

TABLE 15.1

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two Hour Exclusion Boundary (800meters)		Course of Accident Low Population Zone (9600 meters)	
	Thyroid	Whole Body	Thyroid	Whole Body
Accident	(Rem)	(Rem)	(Rem)	(Rem)
Fuel Handling (In Containment)	20	less than 1	1	less than 1

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

The Advisory Committee on Reactor Safeguards completed a partial review of the Diablo Canyon Operating license application in June 1975, and the Committee's report on this partial review was attached as Appendix B to Supplement No. 3 to the Safety Evaluation Report. In its report, the Committee stated that it would complete its review of the seismic design bases, adequacy of the seismic design, protection against tsunamis and certain other matters after the NRC staff's review of seismic-related topics was completed.

The Committee's report also contained other comments and recommendations. The actions we have taken or plan to take in response to these comments and recommendations are described in the following paragraphs.

1. The Committee stated that the results of tests and analyses for 17 x 17 fuel assemblies should be evaluated fully by the NRC staff, and resolved to its satisfaction, prior to the full core use of 17 x 17 fuel to produce power. These included: Fuel assembly flow tests, DNB tests for non-uniform heat flux, and the effect of fuel rod bowing on DNB after the first fuel cycle.

   Our evaluation of the fuel assembly flow tests is completed as discussed in Section 4.2.1 of Supplement No. 3 to the Safety Evaluation Report. Our evaluation of DNB and fuel rod bowing is completed as discussed in Section 4.4 of this supplement.

2. The applicant's discussions with the Committee included a presentation by Westinghouse concerning an augmented startup program proposed for implementation in some of the first Westinghouse reactors with full core employment of the 17 x 17 fuel assembly. The Committee recommended that the NRC staff evaluate the results of the augmented startup program as well as overall operating experience with large, high power-density reactors, prior to sustained operation at full power.

Augmented startup program tests have been performed on two of the first plants using 17 x 17 fuel assemblies, Trojan and Beaver Valley. The licensees have submitted reports on these tests. Further results will be published by Westinghouse in the near future. Preliminary indications are that the core peaking factor varied suprisingly little from the steady state value during load following using constant axial offset control. This is also true for results of tests in 15x15 cores.

As a part of the generic evaluation of 17 x 17 fuel, we insisted on development of the portion of the tests which are designed to show a comparison between calculated and measured power distribution. This position was stated in Section 18.2.3 of Supplement No. 1 to the Safety Evaluation Report for the Beaver Valley Power Station, Unit No. 1 (Docket No. 50-334). It is important to note, however, that while we required these tests, we view them as confirmatory in nature. Thus, we are not

making our evaluation and acceptance of the results of the test program a condition on the further operation of the plants utilizing 17 x 17 fuel. This is consistent with our practice of reviewing startup test reports in parallel with the operation of the plant. We feel this arrangement has proven satisfactory.

3. The Committee stated that the NRC staff should review the effectiveness of the proposed method of constant axial offset control in protecting against adverse consequences of postulated reactor transients and accidents. The Committee wished to be kept informed.

   We considered constant axial offset control in connection with our generic review of 17 x 17 fuel as presented in the Westinghouse Topical Report WCAP-8185, "Reference Core Report 17 x 17," July 26, 1974. We have concluded that this method of control can be an effective method of controlling a reactor so that the maximum peaking factor which occurs in the core will never exceed the maximum allowable. Accordingly, we have found the control method acceptable for use at Diablo Canyon and other reactors now operating or soon to be operated.

   Further, we are performing independent analyses. Our consultants at Brookhaven National Laboratory have completed an independent evaluation of the beginning-of-life analysis of constant axial offset control (A. Burlik et al, "Power Peaking During Load Following Using Westinghouse Constant Axial Offset Power Distribution Control," BNL-NUREG 22477, January 1977). This evaluation confirmed the Westinghouse results. End of life analyses will be performed in the future.

4. The Committee stated that the results of prototype 17 x 17 fuel assembly irradiations should be followed closely.

   Two prototype 17 x 17 fuel assemblies are installed in each of the Surry reactors. Inspection after two fuel cycles in Unit 1 and one fuel cycle in Unit 2 has revealed no anomalies. We will continue to follow the results of these prototype irradiations.

5. The 17 x 17 fuel assemblies to be used in full core implementation of the 17 x 17 fuel design have eight spacer grids, as contrasted with the two prototype 17 x 17 assemblies in each of the Surry reactors (discussed above) which have seven spacer grids. Following each cycle of operation, the 17 x 17 fuel assemblies with eight spacer grids (in the Diablo Canyon reactors and other reactors) will be examined for fuel rod integrity, fuel rod and assembly dimension and alignment and surface deposits. In view of the fact that the 17 x 17 fuel array is a new design and that no prototype irradiations are planned for 17 x 17 fuel containing eight spacer grids, the Committee stated that the results of surveillance programs for this type of fuel should be followed closely. The Committee wished to be kept informed.

   The fuel surveillance program for 17 x 17 fuel assemblies with eight spacer grids is described in further detail in Section 4.2.1 of Supplement No. 2 to the Safety Evaluation Report. We will follow this program closely and, when results become available, we will keep the Committee informed.

6. The Committee wished to be kept informed concerning the results of various ongoing 17 x 17 fuel test and analytical programs, and any design changes which might be proposed in the future.

   Our actions concerning several specific 17 x 17 fuel test and analytical programs are described above. We will keep the Committee informed concerning the results of the various 17 x 17 fuel test and analytical programs and any design changes which might be proposed in the future.

7. The Committee wished to be kept informed concerning the reevaluation of emergency core cooling system performance and operating limits and procedures for power distribution monitoring.

   Our evaluation of this matter is completed as discussed in Sections 4.4 and 6.3 of this Supplement.

8. The Committee stated that the NRC staff review of Anticipated Transients Without Scram should be completed and the matter should be resolved in a manner satisfactory to the NRC staff and to the Committee.

   The status of this matter is described in Section 7.2.5 of this Supplement. We will continue our review and will require that any changes indicated to be needed by the results of approved analyses to be incorporated into the Diablo Canyon plant.

9. The Committee stated that the matter concerning protection against tornado missiles should be resolved in a manner satisfactory to the NRC staff.

Our evaluation of this matter is continuing. We will report the final resolution in a future supplement to the Safety Evaluation Report.

10. The Committee stated that the matter concerning environmental qualification of Class I instrumentation and electrical equipment should be resolved in a manner satisfactory to the NRC staff and the Committee.

Our evaluation of this matter is continuing. We will report the final resolution in a future supplement to the Safety Evaluation Report.

11. The Committee stated that generic problems, relating to large water reactors, should be dealt with appropriately by the NRC staff and the applicant as suitable approaches are developed.

The status of each of these items is discussed in Appendix B to this supplement.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 Preoperational Storage of Nuclear Fuel

In the Safety Evaluation Report we described financial protection and indemnification requirements that the applicant would be required to meet in connection with preoperational storage of nuclear fuel. Since then, the applicant has met these requirements to the extent necessary to obtain fuel storage licenses pursuant to 10 CFR Part 70 and has received such fuel storage licenses for Unit 1 and Unit 2. The applicant will continue to be required to maintain the necessary financial protection and to pay annual fees in connection with the indemnification agreements as described in the Safety Evaluation Report.

21.2 Operating License

In the Safety Evaluation Report we described the financial protection and indemnity requirements for the Diablo Canyon Nuclear Power Station, Units 1 and 2.

By publication in the Federal Register Volume 42, Number 74, April 18, 1977, 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," was amended to increase the amount of primary financial protection from private sources required for facilities having a rated capacity of 100 electrical megawatts or more to \$140 million, effective May 1, 1977. (\$110 million was the figure discussed in the Safety Evaluation Report.) The requirements remain the same as discussed in the Safety Evaluation Report except for the different amount of required protection from private sources.

21.3 Conclusion

On the basis of the above considerations and those identified in the Safety Evaluation Report, we reaffirm our conclusion that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that prior to issuance of any operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to the execution of an appropriate indemnity agreement with the Commission.

22.0 CONCLUSIONS

In Section 22 of Supplement No. 5 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. Resolutions for a number of those items have been presented in this supplement. Items which currently remain outstanding are summarized below.

1. An evaluation of the plant's capability to withstand an earthquake of magnitude 7.5 on the Hosgri fault (Section 3.7 of Supplement No. 4 and Supplement No. 5).

2. An evaluation of the environmental and seismic qualification of Category I electrical, instrumentation and control equipment (Sections 3.10 and 7.8 of the Safety Evaluation Report).

3. An evaluation of the effects of postulated pipe breaks outside containment for Unit 2. However, the evaluation has been completed for Unit 1 (Section 3.6 of this Supplement).

4. An evaluation of the means of protecting the reactor coolant system from overpressurization transients at low temperature for Unit 1 in the long term (after the first fuel cycle) and for Unit 2. However, the evaluation has been completed for the short term provisions for Unit 1 (Section 5.2.2 of this Supplement).

5. An evaluation of the details for implementing industrial security provisions (Section 13.6 of this Supplement).

6. An evaluation of the vulnerability of the electric power systems and equipment to a degraded grid voltage condition (Section 8.0 of this Supplement).

7. An evaluation of a postulated main steam line break inside containment (Section 6.2.1 of this Supplement).

8. An evaluation of the details of modifications to the diesel generator compartment ventilation system (Section 9.5.5 of this supplement).

9. An evaluation of the plant's tornado missile protection (Section 3.5 of Supplement No. 3 to the Safety Evaluation 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 SAFETY REVIEW

September 30, 1976	Letter from applicant providing information about schedule for ATWS analysis requested on July 2, 1977
September 30, 1976	Letter to applicant requesting reevaluation of fire protection capabilities
October 11, 1976	ACRS Subcommittee meeting in Los Angeles, California to discuss seismic design
October 27, 1976	Meeting with applicant to discuss qualification of Class IE electrical equipment
October 27, 1976	Letter from applicant providing schedule for submitting fire protection analysis requested on September 30, 1977
November 13, 1976	ACRS full committee meeting on Diablo Canyon in Washington, D. C. to discuss seismic design
December 3, 1976	Letter to applicant requesting additinal information about containment structural integrity test
December 7, 1976 to December 17, 1976	Public Hearings on Environmental review in San Luis Obispo, California (including routine release of radioactivity - Appendix I to 10 CFR 50)
December 17, 1976	Letter to applicant providing guidance on fire protection reevaluation
December 20, 1976	Letter from Executive Director of ACRS providing comments on seismic design basis
December 28, 1976 and December 29, 1976	Meeting with applicant to discuss seismic design reevaluation
January 5, 1977	Meeting with applicant to discuss seismic design reevaluation and ACRS comments.
January 7, 1977	Submittal of Amendment 46 including (1) changes to quality assurance program, (2) addition of automatic circulating water pump trip and (3) miscellaneous changes

January 10, 1977	Letter to applicant requesting information about reactor coolant system overpressurization protection
January 19, 1977	Letter from applicant submitting report entitled, "Near-Field Ground Motion Simulation for a Vertical Fault with Dip-Slip"
February 4, 1977	Meeting with applicant to discuss seismic design - criteria for reevaluation of structures
February 10, 1977	Letter from applicant about the schedule for submitting information on reactor coolant system overpressurization protection requested on January 10, 1977
February 18, 1977	Letter to House Subcommittee on Oversight and Investigations about Diablo Canyon seismic design licensing situation
February 25, 1977	Letter to applicant providing guidance on complying with new security regulation 10 CFR 73.55
March 11, 1977	Letter to applicant requesting information about postulated fuel handling accident inside containment
March 16, 1977	Letter from applicant providing additional information about containment structural integrity test requested on December 3, 1976
March 18, 1977	Letter to California Energy Commission about the possibility of an interim operating license
March 31, 1977	Letter to House Subcommittee on Energy and the Environment about the Diablo Canyon seismic design licensing situation
March 31, 1977	Submittal of Amendment 47 including (1) information about the qualification of Class IE electrical equipment-including a report on submerged equipment, (2) a reanalysis of ECCS performance with upper head temperature at T(hot) and (3) miscellaneous changes
April 7, 1977	Letter to applicant requesting information about instrument trip setpoint values
April 11, 1977	Letter from PG&E expressing intention to apply for an interim operating license
April 20, 1977	Meeting with intervenor's consultant to discuss Diablo Canyon quality assurance program

April 21, 1977	Submittal of Revision 1 to physical security plan
April 28, 1977	Letter from applicant providing schedule for submitting information about instrument trip setpoint values requested on April 7, 1977
April 29, 1977	Meeting with applicant to discuss seismic design reevaluation-systems needed for safe shutdown
April 29, 1977	Submittal of Amendment 48 including (1) minor revisions to the electric power systems, (2) minor revisions to the meteorology program, (3) information about reactor coolant system overpressurization protection and (4) miscellaneous changes
May 3, 1977	Meeting with applicant to discuss seismic design reevaluation-interim operating license
May 4, 1977	Letter to applicant providing guidance on complying with new security regulation 10 CFR 73.55
May 12, 1977	Prehearing Conference in Los Angeles, California to discuss safety contentiions and hearing schedule
May 23, 1977	Letter to applicant requesting information about inservice inspection program under 10 CFR 50.55a(g)
May 25, 1977	Submittal of Revision 2 to the security plan
May 27, 1977	Submittal of Amendment 49 including (1) minor revisions to tornado protection analysis, (2) minor revisions to information about qualification of Class IE electrical equipment, (3) minor additions to description of ECCS performance evaluation, (4) a detailed analysis of a postulated fuel handling accident inside containment and (5) miscellaneous changes
May 27, 1977	Letter to applicant providing staff comments on Revision 1 to the security plan
June 1, 1977	Letter to applicant requesting information about postulated steam line break inside containment

June 2, 1977	Meeting with applicant to discuss seismic reevaluation and interim licensing
June 3, 1977	Submittal of Revision 3 to the security plan
June 3, 1977	Letter to House Subcommittee on Oversight and Investigations answering additional questions on Diablo Canyon seismic design licensing situation
June 5, 1977	Submittal of Amendment 50 including (1) partial results of seismic reevaluation of the plant, (2) probability studies on earthquakes and (3) other seismic studies and responses to ACRS comments and questions
June 6, 1977	Letter to applicant requesting information about degraded grid voltage
June 7, 1977	Letter to applicant requesting information about postualated pipe breaks outside containment
June 7, 1977	Meeting with applicant to discuss the staff review of the security plan
June 10, 1977	Letter from applicant providing information about ECCS performance evaluation
June 15, 1977	Submittal of Revision 4 to the security plan
June 16, 1977	Letter from applicant providing schedule for submitting information about steam line break inside containment requested on June 1, 1977
June 16, 1977	Letter from applicant revising schedule for submitting information about instrument trip setpoint values requested April 7, 1977
June 21, 1977 to June 23, 1977	ACRS Subcommittee meeting in Los to Angeles, California to discuss seismic design
June 29, 1977	Submittal of Revision 5 to the security plan
June 30, 1977	Congressional hearings on Diablo Canyon seismic design licensing situation-House Subcommittee on Energy and the Environment

July 1, 1977	Letter from applicant providing schedule for submitting information about degraded grid voltage requested on June 6, 1977
July 5, 1977	Letter from applicant submitting informatio about protecting the reactor coolant system from over-pressurization transients
July 6, 1977	Letter from applicant submitting information about postulated pipe breaks outside of containment
July 6, 1977	Letter from applicant submitting information about the tornado wind design of steel siding for the cable spreading rooms and switchgear rooms in the turbine building
July 8, 1977	Letter from applicant submitting information about the analysis of a fuel handling accident inside containment

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS GENERIC ITEMS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic matters applicable to large light-water reactors. The most recent such report was issued on February 24, 1977. In addition, the NRC staff periodically reports on the status of its efforts to resolve these generic items. The latest staff status report was issued on January 31, 1977.

These are items which the Committee and the NRC staff, while finding present plant designs acceptable, believe have potential for adding to overall safety margins and so should be considered for application to the extent reasonable and practicable as solutions are found, recognizing that such solutions may occur after completion of a specific plant. This is consistent with our continuing efforts to reduce the already small safety risk from nuclear power plants.

The current status of each of these generic items as it relates to Diablo Canyon is indicated below. The numbering corresponds to that in the February 24, 1977 report of the Committee.

Group II - Resolution Pending

(1) Turbine Missiles

This item is under generic review. The staff believes that the risk from turbine missiles at Diablo Canyon is small. As part of the generic review we are evaluating the effectiveness of

measures such as frequent valve testing and turbine rotor material inservice inspection in reducing the risk (Staff status report dated January 31, 1977.

(2) Effective Operation of Containment Sprays on LOCA

This item is resolved for Diablo Canyon by use of sodium hydroxide additive to the containment sprays (SER Section 6.2.3).

The generic review continues, including reviews of alternate additives that may be preferable (Staff status report dated January 31, 1977).

(3) Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock

This item is resolved for Diablo Canyon by employment of an acceptable vessel design (SER Section 5.3 and Regulatory Guide 1.2).

The research program is continuing and, as indicated in our status report dated January 31, 1977, the work done to date suggests that flawed, irradiated reactor vessels subjected to thermal shock from LOCA-ECCS water will not fail catastrophically. Further, as indicated in Regulatory Guide 1.2 and Section 5.3 of the SER, if it should be concluded that the margin of safety against reactor vessel brittle failure due to ECCS operation at any time during vessel life is unacceptable, an engineering solution, such as annealing, could be applied.

(4) Instruments to Detect (Severe) Fuel Failures

This item is resolved for Diablo Canyon by employment of a failed fuel detection system (FSAR, Section 7.7.1).

The generic review/research program is continuing as discussed in our status report dated January 31, 1977. The Committee's status report, dated February 4, 1977, indicates that this item is resolved for limited fuel failures but that more work is needed for the severe failure case to establish instrumentation criteria.

(5) Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel

This item is resolved for Diablo Canyon by employment of a loose parts monitor (SER-Section 5.4).

The generic review is continuing and will establish criteria for such systems as indicated in our status report dated January 31, 1977.

(6) Common Mode Failures

This item is under generic review as indicated in our status report dated January 31, 1977.

(7) Behavior of Reactor Fuel Under Abnormal Conditions

A generic research program is underway as indicated in our status report dated January 31, 1977.

(8) BWR Recirculation Pump Overspeed During LOCA

This item is not applicable to Diablo Canyon, which is a pressurized water reactor facility.

(9) The Advisability of Seismic Scram

As indicated in our letter of May 19, 1977 to the Committee, the staff considers the generic studies to be completed and does not plan to require the installation of seismic scram devices on commercial power reactors.

However, the advisability of such a device for Diablo Canyon is still being discussed by PG&E, the NRC staff and the Committee.

(10) Emergency Core Cooling System Capability for Future Plants

This item does not apply to Diablo Canyon, which is an existing plant.

Group IIA Resolution Pending - Items Since December 18, 1972

(1) Control Rod Drop Accident (BWR's)

This item does not apply to Diablo Canyon which is a pressurized water reactor facility.

(2) Ice Condenser Containments

This item does not apply to Diablo Canyon which does not employ an ice condenser containment.

(3) Rupture of High Pressure Lines Outside Containment

We expect this item to be resolved for Diablo Canyon by employment of our current acceptance criteria in our review, which is not yet completed. (Section 3.6 of this supplement).

The staff considers the generic item to be acceptably resolved as indicated in our status report dated January 31, 1977.

(4) PWR Pump Overspeed During a LOCA

This item is under generic review as indicated in our status report dated January 31, 1977 and in Section 5.2.6 of the SER.

(5) Isolation of Low Pressure from High Pressure System

This item is resolved for Diablo Canyon by employment of acceptable design measures. The Residual Heat Removal System overpressurization interlocks are discussed in Section 7.6 of the Safety Evaluation Report. Leak testing of the emergency core cooling system check valves is specified in the ASME Boiler and Pressure Vessel Code, Section XI, and thus will be a requirement of the Technical Specifications (Section 5.2.8 of this Supplement).

The NRC staff considers the generic item acceptably resolved. However, there is a continuing effort including development of an ANSI Standard on the subject (Staff status report dated January 31, 1977).

(6) Steam Generator Tube Leakage

This item is resolved for Diablo Canyon by employment of acceptable measures. These include the use of volatile secondary chemistry

control and titanium condenser tubes to prevent steam generator tube degradation. They also include inspections according to Regulatory Guide 1.83 and monitoring of steam generator tube leakage and secondary system radioactivity to detect any possible degradation in a timely manner.

The generic review is continuing as discussed in our status report dated January 31, 1977.

(7) ACRS/NRC Periodic 10-Year Review of all Power Reactors

This item is under generic review as indicated in our status report dated January 31, 1977.

Group IIB Resolution Pending - Items Added Since February 13, 1974

(1) Computer Reactor Protection System

This item does not apply to Diablo Canyon which does not employ this type of reactor protection system.

(2) Qualification of New Fuel Geometries

This item is resolved for Diablo Canyon by employment of an acceptable fuel design and surveillance program. The evaluation, testing and surveillance for the 17 x 17 Diablo Canyon fuel design are summarized in items 1 through 7 of Section 18.0 of this supplement in response to specific Committee comments in these areas.

The generic review is continuing as discussed in our status report to the Committee dated January 31, 1977. (Also see item II.7 above concerning research into fuel behavior under abnormal conditions).

(3) Behavior of BWR Mark III Containments

This item is not applicable to Diablo Canyon which is a pressurized water reactor facility.

(4) Stress Corrosion Cracking in BWR Piping

This item is not applicable to Diablo Canyon which is a pressurized water reactor facility.

Group IIC Resolution Pending - Items Added Since March 12, 1975

(1) Locking Out of ECCS Power Operated Valves

This item is resolved for Diablo Canyon by employment of acceptable measures which comply with Branch Technical Position EICSB 18 from Appendix 7A to the Standard Review Plan (Section 6.3 of Supplements 4 and 5 to the SER).

The generic review is continuing as indicated in our status report dated January 31, 1977.

(2) Design Features to Control Sabotage

We expect item to be resolved for Diablo Canyon by employing our current acceptance criteria in our review, which is not yet completed (Section 13.0 of this supplement).

The generic review is continuing as indicated in our status report dated January 31, 1977.

(3) Decontamination and Decommissioning of Reactors

This item is under generic review as indicated in our status report to the Committee dated January 13, 1977.

(4) Vessel Support Structures

This item is under generic review as discussed in our status report dated January 31, 1977.

We have concluded that licensing plants for operation is acceptable while our review continues (Section 5.2.1 of Supplement No. 4). In addition, we have requested that the applicant complete the appropriate analyses for Diablo Canyon as soon as possible and we will report on the progress of these analyses in future supplements to the Safety Evaluation Report.

(5) Water Hammer

This item is resolved for Diablo Canyon by employment of acceptable measures to prevent feedwater system water hammer. Diablo Canyon employs a combination of (1) modifications to the feedwater system and steam generators, (2) procedural limitations and (3) testing to ensure the design and procedures will be effective in eliminating feedwater system water hammer (SER Supplement 4, Section 10.4).

The generic studies are continuing as indicated in our status report to the Committee dated January 31, 1977.

(6) Maintenance and Inspection of Plants

This item pertains to design improvements for future plants. Accordingly, it does not apply to Diablo Canyon which is an existing plant.

(7) Behavior of BWR Mark I Containments

This item does not apply to Diablo Canyon which is a pressurized water reactor facility.

Group IID Resolution Pending - Items Added Since April 16, 1976

(1) Safety Related Interfaces Between Reactor Island and Balance-of-Plant

This item pertains to standard plants. Accordingly, it does not apply to Diablo Canyon which is not a standard plant.

(2) Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

This is a new item that was not addressed in our status report dated January 31, 1977. The generic matter will be discussed in a future staff status report.

Our evaluation of the qualification of Class IE electrical, instrumentation and control equipment for Diablo Canyon is not yet completed (SER Sections 3.10 and 7.8). However, the proper design and qualification testing of such equipment is included in our evaluations and will be considered. The Quality Assurance program for Diablo Canyon provides for the use of approved procedures and checks in performing maintenance on such equipment. We have approved the quality assurance program (Section 17.0 Safety Evaluation Report Supplement No. 3).

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Docket Nos. 50-275
and 50-323

APPLICANT: Pacific Gas & Electric Company (PG&E)

FACILITY: Diablo Canyon Nuclear Power Station, Units 1 and 2
(Diablo Canyon)

SUMMARY OF MEETING HELD ON JUNE 2, 1977, TO DISCUSS DIABLO CANYON SEISMIC DESIGN

We met with the applicant on June 2, 1977, to discuss the seismic design of Diablo Canyon. A list of attendees is provided in the enclosure.

BACKGROUND

Diablo Canyon had originally been designed to withstand an earthquake with a horizontal ground acceleration of 0.4g, based on the geological investigations that had been conducted in connection with the construction permit review. During the operating license review, which was in progress, we had requested that PG&E reevaluate the plant's seismic capabilities to determine what modifications might be necessary to ensure that the plant could withstand a more severe earthquake with a horizontal ground acceleration of 0.75g, based on newer geological information. PG&E was performing such a reanalysis.

PG&E had also expressed its intention to apply for an interim operating license for Diablo Canyon to allow plant operation while the seismic reevaluation was being completed. The material needed to justify such an interim license application had previously been outlined by the NRC staff as follows:

(1) A demonstration of the need to consider such an action, and

(2) Information and analyses to demonstrate that the requisite level of safety would be assured during the period of the interim license. This information should include all available results of the seismic reassessment program, supported by:

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(a) a realistic assessment of the probability of large earthquakes in the site environs and the probability of the plant to withstand such earthquakes without failures of structures and equipment sufficient to lead to unacceptable radiological consequences to the public;

(b) a commitment to make any changes to the design determined to be necessary on the basis of the continuing seismic reassessment program; and

(c) an evaluation of the practicality of making the need changes to a plant which has been in operation during the term of the interim license.

INTERIM LICENSE APPLICATION - RELATIONSHIP TO REGULATIONS

At a previous meeting on May 3, 1977, we had discussed the question of whether or not the interim license application would constitute a request for an exemption to the NRC Regulations, in particular to Appendix A to 10 CFR Part 100 and to Criterion 2 of the General Design Criteria (Appendix A to 10 CFR Part 50).

At this meeting, June 2, 1977, we informed PG&E of our opinion on this subject as follows:

(1) Regardless of whether or not the interim license was to be considered an exemption from, an exemption to or a waiver of the regulations, the information needed to support the application, outlined above, would be substantially the same. The fundamental criterion in any approach would be that the plant must be shown to have an acceptable level of safety before an operating license would beissued.

(2) We had considered three possible ways of stating the interim license application:

(a) the first possibility would be a petition for an exemption or waiver of a particular rule under 10 CFR Part 2.758 (b). The sole basis that would be allowed here was that due to special circumstances application of the particular rule would not serve the purpose for which the rule was intended. In this case, the regulations explicitly spelled out the subsequent procedures to be followed by the Licensing Board.

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(b) The second possibility would be request for an exemption from the requirements of the regulations under 10 CFR Part 50.12. Here the regulations indicated that the requested exemption would have to be shown to:

(i) be authorized by law,

(ii) not endanger life or property or the common defense and security, and;

(iii) be otherwise in the public interest.

In this case, the subsequent procedures to be followed by the Licensing Board were not explicitly spelled out in the regulations.

(c) The third possibility would be a showing that the requested action would not constitute a deviation from the regulations in that the pertinent regulations already contemplated and allowed for an applicant to propose and justify alternate approaches. Language to this effect was contained in Appendix A to 10 CFR Part 50, 10 CFR Part 100 and Appendix A to 10 CFR Part 100.

(3) In any of these methods the overriding and fundamental consideration would be whether or not an acceptable level of safety had been shown.

(4) In any of these methods we would want any specific passage or section of the regulations that might not be met to be clearly identified and the reasons and justifications to be clearly stated.

(5) We indicated that 10 CFR Part 50.57, "Issuance of operating license", would be cited in connection with any of the three methods discussed above.

(6) The staff would not specify which method of stating the application should be used. This decision would be left to PG&E.

INTERIM LICENSE APPLICATION - STATUS OF SYSTEMS REANALYSIS

At a previous meeting, on April 29, 1977, PG&E had described to us those systems and portions of systems for which they intended to complete the reanalysis prior to applying for an interim license. In general, these consisted of the systems that PG&E considered necessary to ensure that the plant could be safely shutdown following a major earthquake. At this

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meeting, on June 2, 1977, we informed PG&E that we considered the probability study to be the primary tool that would be used in judging whether or not the plant would have an acceptable level of safety for an interim operating license. Accordingly, we did not consider that the interim license would be contingent upon whether or not the reanalysis had been completed for any particular system or part of a system at the new earthquake level of 0.75g.

SEISMIC REEVALUATION - COMBINATION OF LOADS

In order to justify a full term operating license, PG&E would need to complete the seismic reevaluation at 0.75g. We informed PG&E that in performing this reevaluation they should combine the calculated loads resulting from a postulated loss-of-coolant accident with the calculated loads resulting from the postulated earthquake at 0.75g. These loads should be combined by direct addition, as is the usual practice in nuclear plant design, rather than by using the square root of the sum of the squares. PG&E would then be expected to demonstrate to us that structures, systems and components important to safety can perform their required safety functions under the combined loading conditions. If, for any particular item, functional capability could not be demonstrated and PG&E should believe that modifications to demonstrate functional capability would be impractical or unwarranted, then PG&E would be expected to describe the situation fully and to justify its acceptability.

Original Signed By
Dennis P. Allison
Dennis P. Allison, Project Manager
Light Water Reactors Branch No. 1
Division of Project Management

Enclosure:
Attendance List

cc w/enclosure:
See Page 5

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DATE	6/29/77	6/29/77				

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ENCLOSURE

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JUNE 2, 1977

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P. Crane
J. Hoch
J. Gormly

WESTINGHOUSE

W. Gangloff
T. Esselman

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J. Stolz
D. Allison
J. Knight
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CENTER FOR LAW IN THE PUBLIC INTEREST
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D. Fleishaker

-76) NRCM 0240

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MEETING SUMMARY

Docket File
NRC PDR
Local PDR
TIC
NRR Reading
LWR 1 File
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