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December 15, 2000

PG&E Letter DCL-00-157

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-323, OL-DPR-82
Diablo Canyon Unit 2
Response to Request for Additional Information Regarding License Amendment Request 00-04, Revision of Technical Specification 3.5.2 – Increase in Charging Pump Completion Time During Unit 2 Cycle 10 from 72 Hours to 7 Days

Dear Commissioners and Staff:

In PG&E Letter DCL-00-086, "License Amendment Request 00-04, Revision of Technical Specification 3.5.2 - Increase in Charging Pump Completion Time During Unit 2 Cycle 10 from 72 Hours to 7 Days," dated June 2, 2000, PG&E submitted License Amendment Request (LAR) 00-04 to amend the facility operating license for Diablo Canyon Power Plant (DCPP) Unit 2 to increase the Centrifugal Charging Pump (CCP) 2-1 completion time to 7 days during cycle 10. The change will allow for a potential on-line repair or replacement of CCP 2-1.

During a conference call on October 30, 2000, PG&E discussed NRC questions regarding the Probabilistic Risk Assessment (PRA) input that supports LAR 00-04. The NRC staff requested clarification in the following areas:

1) discussion of the PRA model reflection of the as-built and current plant design and a summary of the PRA updates since NRC approval,
2) discussion of the peer review process for the PRA model,
3) the independence of the internal review of the PRA model,
4) the quality assurance program for the PRA model, and
5) results of reviews of the accident sequences for loss of a charging pump.

PG&E has developed and maintained a PRA program over the last 12 years for application at DCPP. It was originally reviewed by the NRC and found to be "beyond the state of the art" by the NRC after review as part of the Long Term Seismic Program at DCPP in 1988. Since that time, PG&E has performed several updates to the PRA to assure the model is consistent with the as-built and operated plant. Additionally, the NRC has reviewed both the DCPP individual plant examination (IPE) and individual plant examination of external

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events (IPEEE). The background and summary of the model, including the PRA model reflection of the as-built and current plant design and the PRA updates since NRC approval, is provided in the Diablo Canyon PRA summary in Enclosure 1.

A peer review by the Westinghouse Owners Group (WOG) in May 2000 confirmed that PG&E has maintained a high quality PRA that is appropriate for risk-informed submittals. The peer review team evaluated the PG&E independent review process implemented in the PRA program calculations and found it to be high quality. The PRA program calculations receive a similar level of independent review as the design calculations at DCPP receive, and are performed in accordance with plant procedures. Enclosure 2 provides a discussion of the WOG peer review results, the PG&E program for independent review, and the quality assurance program for the PRA.

The results of reviews of the accident sequences for loss of a charging pump are provided in Enclosure 3. The accident sequences are dominated by medium loss-of-coolant accident scenarios.

If you have additional questions regarding the PRA evaluation used in support of LAR 00-04, please contact Mr. Ken Bych at (805) 545-4241.

Sincerely,

Lawrence F. Womack

cc: Edgar Bailey, DHS
Girija Shukla
Ellis W. Merschoff
David Proulx
Diablo Distribution

Enclosures
1. Diablo Canyon PRA Summary
2. Summary Discussion of WOG Peer Review Process and Results
3. Sequences with CCP 2-1 Out of Service

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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|---|---|
| In the Matter of<br>PACIFIC GAS AND ELECTRIC COMPANY<br><br>Diablo Canyon Power Plant<br>Unit 2 | Docket No. 50-323<br>Facility Operating License<br>No. DPR-82 |

## AFFIDAVIT

Lawrence F. Womack, of lawful age, first being duly sworn upon oath says that he is Vice President, Power Generation and Nuclear Services, Pacific Gas and Electric Company; that he is familiar with the content thereof; that he has executed the additional information regarding License Amendment Request 00-04 on behalf of said company with full power and authority to do so; that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Lawrence F. Womack
Vice President, Power Generation and Nuclear Services

Subscribed and sworn to before me this 15th day of December, 2000
County of San Luis Obispo
State of California

Notary Public

LISA M. NELSON
Commission #
Notary Public - California
San Luis Obispo County
My Comm. Expires

# DIABLO CANYON PRA SUMMARY

## INTRODUCTION

This enclosure provides a summary of the Diablo Canyon Power Plant (DCPP) Probabilistic Risk Assessment (PRA) model. It provides justification that the model reflects the as-built and current plant design and provides a summary of the updates to the model since approval.

The models summarized in this enclosure represent an enhancement to the original Diablo Canyon Probabilistic Risk Assessment (DCPRA-1988) (Reference 1) performed as part of the Long Term Seismic Program (LTSP) (Reference 2). The LTSP reevaluated the seismic design bases for DCPP, as specified in the Unit 1 Full-Power Operating License, DPR-80, Condition 2.C.(7). As part of the LTSP, PG&E was required by the license condition to complete "a probabilistic risk analysis and deterministic studies, as necessary, to assure adequacy of seismic margins." To meet this requirement, the DCPRA-1988 was completed in 1988. The DCPRA-1988 is a full-scope Level 1 PRA that evaluated the probable frequency of experiencing reactor and plant damage resulting from internal and external initiating events. While it was performed for DCPP Unit 1, the DCPRA-1988 is equally applicable to DCPP Unit 2 because of the substantial similarities between the two units. The NRC reviewed the LTSP and issued Supplement No. 34 to NUREG-0675 (Reference 3) in June 1991, accepting the DCPRA-1988. The DCPRA-1988 was reviewed for the NRC primarily by Brookhaven National Laboratory (BNL). This review is documented in NUREG/CR-5726 (Reference 4).

To fulfill the requirements of NRC Generic Letter (GL) 88-20 (Reference 5), and as part of PG&Es ongoing risk management tasks, PG&E updated the DCPRA-1988 for the individual plant examination (IPE) submittal. This was transmitted to the NRC on April 14, 1992 in PG&E Letter DCL-92-087, "Response to Generic Letter 88-20, Individual Plant Examination."

To fulfill the requirements of GL 88-20, Supplement 4, PG&E completed an individual plant examination of external events (IPEEE) and submitted it to the NRC on June 27, 1994 in PG&E Letter DCL-94-133, "Response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events for Severe Accident Vulnerabilities."

## OVERALL METHODOLOGY

The DCPRA model uses a method that closely follows the series of analytical tasks and methods developed by Pickard, Lowe, and Garrick (PLG) and implemented in performing more than 20 full-scope and phased PRAs of U.S. and foreign nuclear power plants. This is commonly referred to as the "large event tree, small fault tree" methodology. The original exposition of the theoretical and mathematical bases for the approach is provided in the PLG methodology document (Reference 6).

**LIVING PRA**

As part of maintaining a living PRA, the model is periodically updated. The following describes the updates to the original DCPRA model.

DCPRA -1988 Model

The DCPRA -1988 model was the original model developed for the Long Term Seismic Program as part of a condition of licensing Diablo Canyon (references (1) and (2)). It was performed by both PG&E personnel and its consultants, mainly PLG. The model was on PLGs main frame computer system. It was reviewed by the NRC staff and its consultants BNL as documented in references (3) and (4).

DCPRA - 1990 Model

The DCPRA - 1990 model was a transition model. The purpose of this model was to migrate from PLGs main frame computer system model to RISKMAN on the PC, and compare the results with the DCPRA 1988 model. This was the start of model documentation under PG&Es procedures which required verification of each calculation file by an independent reviewer.

DCPRA - 1991 Model

The DCPRA - 1991 model was used to support PG&Es response to GL 88-20. Internal flooding and containment performance (level II) were added to the model. The NRC completed their evaluation of the DCPP IPE June 30, 1993, which included a review of the Level 2, by its contractor Scientech, Inc., SCIE-NRC-210-92.

DCPRA - 1993 Model

The DCPRA - 1993 model was used to support PG&E's response to GL 88-20, supplement 4. It updated the PRA database for plant design and operational data through December 31, 1991. The seismic, fire, and internal PRA models from the DCPRA were updated. The NRC completed their evaluation of the DCPP IPEEE December 4, 1997, which included a Step 1 review of the IPEEE by its contractor Energy Research, Inc. (ERI) , ERI/NRC 95-503.

DCPRA -1995 Model

The DCPRA - 1995 model updated the 1993 model based on plant-specific information on component reliability and unavailability data through December 31, 1994. Additionally, the model was updated based on the plant hardware and procedural changes. Industry events and PRA staff comments and observations about the previous model were incorporated. This was the first model that was used to support online risk assessments for the Operations department.

DCPRA - 1997 Model

During the DCPRA – 1997 model update, two auxiliary tasks were performed in addition to the update activities performed for the previous model. The control room fire scenarios were reevaluated and a Large Early Release model was generated. The online risk configurations were further refined to support the online risk assessments performed by the Operations department. The internal sections of this model were used for risk information in support of the one time completion time extension for centrifugal charging pump 2-1, submitted in PG&E Letter DCL-00-086, dated June 2, 2000.

DCPRA - 00 Model (ongoing)

The major activities for the DCPRA – 00 model update are the internal and seismic models, and generation of the model evaluated by the Westinghouse Owners Group peer review process. This update is continuing currently.

## SUMMARY OF RESULTS

In the DCPRA -1997 model, the core damage frequency (CDF) and the large early release frequency (LERF) figures of merit, due to internal initiating events (including flood events), were estimated to be 3.3 X $10^{-5}$ per year and 8.9 X $10^{-7}$ per year, respectively (point estimate value). The CDF due to seismic events was estimated to be 3.7 X $10^{-5}$ per year.

Since there are uncertainties in the initiating event frequencies, component failure rates, and equipment maintenance unavailability, the uncertainty in the CDF and LERF figures of merit is also analyzed but not presented here.

## REFERENCES

1. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific Gas and Electric Company, PLG-0637, July 1988.

2. Pacific Gas and Electric Company, "Long Term Seismic Program Final Report," PG&E Letter No. DCL-88-192, July 31, 1988.

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report," NUREG-0675, Supplement No. 34, Docket Nos. 50-275 and 50-323, June 1991.

4. Bozoki, G., et al., "Review of the Diablo Canyon Probabilistic Risk Assessment," NUREG/CR-5726, published August, 1994.

5. U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, November 23, 1988.

6. S. Kaplan, G. Apostolakis, B.J. Garrick, D.C. Bley, and K. Woodard, "Methodology for Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0209, June 1981.

## SUMMARY DISCUSSION OF WOG PEER REVIEW PROCESS AND RESULTS

This enclosure summarizes the peer review process, independent review process, and Quality Assurance (QA) program for the Diablo Canyon Power Plant (DCPP) Probabilistic Risk Assessment (PRA) model.

## PROCESS OF WOG PEER REVIEW

A Westinghouse Owners Group (WOG) peer review was conducted in May of 2000 at DCPP. The general assessment of the peer reviewers was that the DCPP PRA can be effectively used in risk-informed applications and licensing submittals to the NRC.

PG&E received strengths in the areas of completeness of initiating events, level of detail of accident sequences in plant modeling, coverage of common cause in the model, interactions of PRA group with plant staff, and application of PRA to support plant operations, and the conduct of a detailed self assessment.

PG&E received input from the WOG peer reviewers that the human reliability analysis area was in need of further evaluation. It was categorized as outdated. In response, PG&E immediately addressed the outdated calculation by contracting Scientech in August 2000 to perform a new calculation. That calculation was completed in October 2000 and is being implemented in the current revision of the model. At this time, it appears that the original results were conservative. PG&E is also changing its administrative procedure to require updating of the human reliability dependency analysis as a routine part of quantification. These actions have addressed the recommendations from the peer review.

Any time a risk-informed application is made, PG&E evaluates the quality of the model for the particular application. Each of the qualitative aspects of the WOG peer review areas have been reviewed against the License Amendment Request 00-04, submitted in PG&E Letter DCL-00-086, dated June 2, 2000, and have been determined to be of no impact.

The peer review determined that 10 of the 11 reviewed areas met the standard for risk-informed submittals or better. Following the changes PG&E made to the eleventh reviewed area, human reliability analysis as described above, the eleventh area now meets the same standard.

Additionally, the peer review correctly identifies PG&E as using the Westinghouse reactor coolant pump (RCP) seal loss-of-coolant (LOCA) model with plant specific MAAP analyses. PG&E has high temperature seals installed at DCPP. PG&E is following the discussions that the WOG and NRC are conducting on the comparisons between the Westinghouse model and the Rhodes model and is supporting the WOG in reaching agreement with the NRC on a modified Rhodes model for use within the WOG and at DCPP.

Lastly, an important aspect of the PRA model is the plant specific information relative to the PRA model. Although DCPP is comparable to other 1970s vintage Westinghouse 4 loop pressurized water reactors, there are specific design and operational capabilities that impact the plant's risk profile. These are listed in Table 2-1.

**QUALITY STANDARDS FOR DCPP PRA CALCULATIONS**

PG&E applied design basis configuration control processes and procedures to PRA calculations. PG&E Nuclear Power Generation Procedure CF3.ID15, "Development and Independent Verification of Calculations or Computer Programs," is the governing procedure for PRA calculations. Within this procedure are requirements to conduct independent verification of calculations. The application of this process was a specific review item addressed by the WOG peer review team in May 2000. That review found the independent verification process conducted by PG&E met the standards for a risk-informed submittals or better.

PG&E controls PRA model software consistent with NPG software QA program. NPG procedure CF2.ID2, "Software QA," applies to applications of "Riskman."

**TABLE 2-1**
**DIABLO CANYON POWER PLANT PRA - SELECTED UNIQUE PLANT & PROCEDURAL FEATURES THAT MAY AFFECT THE RISK PROFILE**

| Design Features | Potential Impact |
|---|---|
| 3 Emergency Diesels Generators (EDGs), 3 Vital Buses, 2 Trains | Advantage for power redundancy. Failure of one EDG does not fail a complete train. |
| EDGs are air cooled. | No dependency on auxiliary salt water (ASW) - vital service cooling water. |
| Can back-feed from 500 KV remotely. | Recovery from loss of power to the switchyard which is the preferred source of power. |
| Can cross-tie vital buses, so one EDG can feed two vital buses. | Greater redundancy in case one EDG fails and a redundant component on another bus fails. |
| Can use other units EDG to feed vital or nonvital buses on the affected unit. | Unit 2 EDGs can be used to support Unit 1 in a Unit 1 station blackout scenario. |
| Offsite power comes from two different switchyards, which are fed from two different transmission systems. | Lower chance of loss of offsite power due to loss of one switchyard or one offsite power transmission system. |
| Two EDG fuel oil trains feed both units, each train has two sources of power. | Per the initial PRA, installed a third (portable) pump for extra redundancy. |
| Have five battery chargers to feed three DC trains. | Have two installed spare battery chargers that can be lined up in about 10-15 min. |
| Proceduralized backup connection of Fire Water cooling to charging pump lube oil. | On loss of component cooling water/ASW can still provide cooling to charging pump(s) to maintain reactor coolant pump seal injection, with operator action. |
| ASW can be cross-connected from the control room. | Have four ASW pumps that can supply both units. |
| Have three power operated relief valves (part of the "full load rejection" design capability) | Only two needed for feed and bleed |
| AMSAC is installed but not modeled in the PRA. | Causes a heavier interaction between auxiliary feedwater (AFW) and solid state protection system in general transient sequences. (Being added to the present model.) |
| Residual Heat Removal (RHR) has one common suction line from reactor coolant system for both trains. | Failure of either bus 'G' or 'H' will disable RHR closed loop cooling. |
| AFW has multiple sources of water | Redundant sources for AFW. |

| Design Features | Potential Impact |
|---|---|
| Little heating, ventilation, or air conditioning is needed due to air cooled equipment and near ocean environment. | Reduced support system failure impact on frontline systems. |
| Remote location | Few people affected by a release. |

## SEQUENCES WITH CCP 2-1 OUT OF SERVICE

This enclosure provides the accident sequences for loss of the centrifugal charging pump (CCP) 2-1. The increase in core damage frequency (CDF) with CCP 2-1 out of service is about 1.45E-5/yr. Of that, 9.8E-6/yr is from internal events and 4.7E-6/yr is from seismic events. The Diablo Canyon Power Plant (DCPP) success criteria for high head injection is 1 out of 4 pumps (CCPs and safety injection (SI) pumps) for small loss-of-coolant accident (LOCA) and 2 out of 4 pumps for medium LOCA (MLOCA).

The increase in the internal model is mostly due to five MLOCA sequences. Four of the five sequences are the probabilistic failure of Train 'B' of the solid state protection system, which in turn fails CCP 2-2 and SI Pump 2-2. In the proposed configuration, CCP 2-1 is cleared, leaving one SI pump available for injection, which is not sufficient for MLOCA. The fifth sequence is the probabilistic failure of both SI pumps, leaving only CCP 2-2 available for injection.

In the seismic model there is an additional conservatism inherent in the calculation of risk of removing one CCP from service. Based on comments from Brookhaven National Laboratory, and as part of the Individual Plant Examination response, PG&E added the conservative assumption that there would always be a seismically induced very small LOCA (VSLOCA) or leak in the reactor coolant system. This could only be mitigated by makeup supplied by a CCP and not an SI pump (i.e. no credit for operator action to decrease RCS pressure so that the SI pumps could supply injection with their 1500 psi discharge pressure).

In the top 100 sequences of the combined internal and external model, 12 additional seismic sequences were identified with CCP 2-1 out of service for 3.1E-6/yr (about two thirds of the total seismic increase). All twelve of these sequences were for a seismically induced VSLOCA, in which SI pumps were not credited for injection. If the SI pumps had been credited, there would have been no new seismic sequences in the top 100 sequences.