ENCLOSURE 2

SCIE-NRC-210-92

TECHNICAL EVALUATION REPORT OF THE DIABLO CANYON INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

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1. INTRODUCTION

This technical evaluation report (TER) documents the results of the SCIENTECH submittal only review of the Diablo Canyon Individual Plant Examination (IPE) Back-End submittal[1], based on the submittal only review objectives set forth by the U.S. Nuclear Regulatory Commission. These review objectives include the following:

- To determine if the IPE submittal provides the level of detail requested in the "Submittal Guidance Document," NUREG-1335
- To assess the strengths and the weaknesses of the IPE submittal
- To pose a preliminary list of questions about the IPE submittal, based on this limited submittal only review
- To complete the IPE Evaluation Data Summary Sheet.

This technical evaluation report complies with the requirements of the contractor task order for submittal only review. Section 2 of the TER provides a summary of our findings, and a brief description of the Diablo Canyon IPE submittal as it pertains to the work requirements, as outlined in the NRC contractor task order. Each portion of Subsection 2.1 corresponds to a specific work requirement. Subsection 2.2 sets out the SCIENTECH assessment of IPE submittal strengths and weaknesses. Subsection 2.3 contains SCIENTECH questions and requests for information from the IPE authors. In Section 3, the IPE is evaluated overall and conclusions are drawn based on the submittal only review. Appended to this report is the completed IPE Evaluation Summary Sheet.

2. CONTRACTOR REVIEW FINDINGS

2.1 Review and Identification of IPE Insights

This section follows the structure of Task Order Subtask 1.

2.1.1 General Review of IPE Back-End Analytical Process

2.1.1.1 Completeness

The IPE submittal is essentially complete with respect to the level of detail requested in NUREG-1335.

2.1.1.2 Description, Justification, and Consistency

The IPE methodology used is clearly described and its selection is justified. The approach followed is consistent with Generic Letter GL 88-20, Appendix 1.

2.1.1.3 Process Used for IPE

The IPE reviewed by SCIENTECH was an extension of the Diablo Canyon PRA (DCPRA) performed by PLG in 1988. DCPRA-1988 was a full-scope Level 1 PRA, conducted to support the Long Term Seismic Program (LTSP)[2]. PG&E retained PLG to help the PRA team perform the Level 2 portion of the IPE. The IPE containment response analysis made use of information from the Zion plant analysis, which is presented in References 3 and 4 of this report. Out of 72 top events identified in Volume 7 of NUREG/CR-4551 for the Zion Accident Progression Event Tree (APET), 30 top events were selected for DCPP CET. These events are listed in Table 4.5-2 of the IPE submittal. The DCPP containment event tree is shown in Figure 4.5-1, page 4.5-15, of the same submittal. The bases for split fraction values used in the quantification of the CET were derived from DCPP-specific analyses and expert opinion values from the Zion plant analysis[4].

2.1.1.4 Peer Review of IPE

The IPE peer review process is discussed in Subsection 5.2 and Table 5-3 of the submittal illustrates the PG&E review. In addition to the in-house review by the PG&E staff and its contractor, PLG, two independent teams of consultants reviewed the IPE. They were the IPE Partnership (IPEP) and the Electric Power Research Institute (EPRI). According to the authors of the IPE submittal, the IPEP concluded that the current Diablo Canyon characterizations of severe accident phenomena and containment failure modes are conservative. EPRI concluded that the IPE source term analysis is conservative. These findings appear on page 5-5 of the submittal.

Table 5-5 of the submittal lists the specific comments of PG&E, based on the in-house review, and the utility's response. Table 5-6 does the same with respect to the comments of PLG and the utility's response to them.

A reference is made in Table 5-5 to an EPRI comment about accident progression analysis. EPRI recommended that, whenever possible, simulation should be carried out to basemat penetration, and a sensitivity study should be performed to evaluate core coolability.

In addressing EPRI's comment, the IPE report states on page 5-6:

> PG&E extended several simulations through to containment basemat penetration and found that the penetration times for other accident sequences could be estimated. Additionally, a sensitivity study was performed to evaluate the importance of core debris coolability with respect to the prevention of concrete ablation.

The peer review process appears to conform with the Generic Letter guidance.

2.1.2 Containment Analysis/Characterization

2.1.2.1 Front-end Back-end Dependencies

The transition from the front-end analysis to the back-end analysis was accomplished by binning each Level 1 core damage sequence into one of many plant damage states as discussed in Subsection 3.16 of the IPE submittal. To reduce the PDSs to a manageable number, the PDSs with negligible frequency were grouped with those of higher frequency and higher consequences. The authors of the IPE referred to this process as "conservative condensation." PDSs were subsequently rebinned into key plant damage states (KPDSs). A set of general guidelines was employed along with engineering judgment to define a set of 16 KPDSs. The guidelines are listed on pages 4.3-2 to 4.3-3. In general, the lower frequency PDSs were combined into higher frequency PDSs with more severe consequences.

As described in Subsection 4.3 of the IPE submittal, in each of the KPDS one or more representative sequences were simulated using MAAP. The entry conditions to CET were made through KPDSs, as discussed in Subsection 4.6. The analysis appears to have taken proper account of front-end to back-end dependencies, and is logically and clearly presented.

2.1.2.2 Sequence with Significant Probability

Table 4.8-1 of the submittal lists the top 100 containment failure sequences. A summation of the frequencies of these sequences is 6.6E-5, which is about 74 percent of the total CDF (8.8E-5). Page 4.8-1 of the submittal notes that the top 100 containment

failure sequences constitute 93 percent of the containment failure frequency. Therefore, 80 percent of the total CDF may lead to some sort of containment failure mode release category. That is, the probability that containment is intact is only 20 percent.

In Table 4.8-2 of the submittal, the frequency of core damage sequences that result in long-term, intact containment is 3.64E-05, which is 41.4 percent of the total CDF. The reason for this discrepancy is that the IPE authors included 52 percent of the long-term overpressurization sequence frequency in the long-term containment intact release category group. The authors of the IPE argued that the likelihood exists to recover containment heat removal systems after 48 hours, and thereby maintain containment integrity.

2.1.2.3 Failure Modes and Timing

The containment failure characterization is described in Subsection 4.4 of the IPE submittal. The important postulated failure modes and leak areas with associated variabilities were estimated, as shown in Tables 4.4-1 and 4.4-2. The containment overpressurization fragility was analyzed probabilistically by considering the following failure modes:

- Cylindrical wall failure in the hoop direction
- Cylindrical wall failure in the meridional direction
- Dome failure
- Personnel hatch failure
- Equipment hatch failure
- Shear failure cylindrical wall at the base
- Base slab failure
- Containment/auxiliary building interaction
- Fuel transfer tube failure
- Major penetration failure.

The median capacity associated with each potentially significant failure mode varied. The lowest median is listed for containment/auxiliary building interaction failure mode, which is 147 psig (Table 4.4-1).

Table 1 summarizes key design features of the Diablo Canyon plant and containment systems.

Table 1 Summary of Key Plant and Containment Design Features of Diablo Canyon

Design Feature	Data
Power Level (MW_{ui}) Unit 1/Unit 2	3,338/3,411
Volume of RCS Water, m^4	310
Mass of Fuel, ton	101
Mass of Zr, ton	21
Containment Free Volume, m^4	74,500
Design Pressure, psig	47
Failure Pressure (median), psig	147
Containment Free Volume/Power, m^4/MW_{ui}	22
Mass of H_3 Generated by Zr Oxidation, Kg	936

2.1.2.4 Containment Isolation Failure

The containment isolation failure and subsequent 480V power failure are described in the submittal as containing approximately 7 percent of the core damage frequency. The dominant Level 1 sequences leading to containment failure mode are described as follows:

- 480V switchgear ventilation failure and subsequent 480V power failure. As a result, the RCP seal return line is not isolated.
- SSPS Train A and B failures. As a result, no containment isolation signal is given.
- Loss of control room ventilation. This leads to SSPS failure and subsequently to failure of containment isolation signal generation.
- Loss of buses G and H. This precludes control room isolation of the seal return line.

2.1.2.5 System/Human Response

Examples of the IPE integrating system/human response with the phenomenological aspects of accident progression into CETs are given as follows:

- Taking credit for operator recovery of containment heat removal systems 48 hours after initiation of the event. The likelihood of this recovery action is noted as 90 percent. The implication of this recovery action is to arrest 32 percent of long-term containment overpressurization accidents (see Table 2 of this report).

- Activating core cooling: CET Top Event AD - Failure to Arrest Core Damage and Prevent Vessel Breach (page 4.5-3). Based on the split fraction values given in Table 4.6.3-1, page 4.6-55, the probability of recovery actions has values up to 26 percent.

2.1.2.6 Radionuclide Release Characterization

Radionuclide release categories are discussed in Subsection 4.7 of the IPE submittal. The following six items, addressed in the definitions of the PDSs, the CETs, or both, were taken into consideration in defining the release categories (see Subsection 4.7.1):

- Containment bypass
- RCS pressure at vessel failure
- Time of containment failure
- Size of containment failure
- Containment spray system
- Debris coolability.

In Table 4.7-1 of the IPE submittal, 37 release categories are listed: 32 are listed for containment failures (RC01 and RC01U through RC16 and RC16U; the letter "U" appears after the names of coolable debris categories to signify noncoolable debris categories); the remaining five categories are for the following: SGTR (RC17); interfacing systems LOCA (RC18); nonsevere core damage sequence (RC19); long-term containment intact sequence (RC20); and basemat melt-through sequence.

Two computer codes, ZISOR and MAAP, were used to generate the source terms for each release category. As noted in Subsection 4.7.2.2, page 4.7-6, of the IPE submittal, the MAAP code was forced to agree with the CET release category being modeled by manually imposing a containment failure, either small or large, in the MAAP simulation at the time required to match the release category definition. Eight of the 37 release categories (RC05U, RC07U, RC09, RC09U, RC11, RC11U, RC13U, and RC15), had frequencies of less than 1E-10/year. Table 4.7-4 of the IPE submittal lists the release fractions of the remaining 29 categories as calculated from ZISOR and MAPP. The leading source term category in terms of frequency of release was Small-Late CF, which

had a frequency of 1.98E-05 /year. A comparison of ZISOR and MAAP results appears in Subsection 4.7.2.3.1 of the IPE submittal.

In Table 4.7-4, the following important characteristics of release categories are not listed:

- Time of release
- Duration of release
- Energy of release
- Height of release
- Warning time for evacuation.

Five representative release timings for Small, Early (RC14); Large, Early (RC04); Small, Late (RC10); Large, Late (RC06); and Bypass (RC17) containment releases are given in Subsection 4.7.2.3.2, page 4.7-8, of the IPE submittal. Insights from the source term analysis are discussed in Subsection 4-7.2.4, page 4.7-9. Containment sprays, it is noted, played a minor role in the overall source term evaluation, although they are very effective in reducing source terms.

In summary, the submittal does appropriately document the radionuclide release characterizations for accident sequences exceeding the screening criteria.

2.1.3 Quantitative Core Damage Estimate

2.1.3.1 Severe Accident Progression

As noted in Subsection 4.6.1, page 4.6-1, of the IPE submittal, the MAAP computer code was used to simulate accident progression for a representative sequence in each of the 16 KPDSs. MAAP model parameters selected by the user provided a framework in which to analyze phenomena, such as the amount of hydrogen produced in-vessel, the ex-vessel debris coolability, the degree of fragmentation of the core debris at the vessel failure, and the magnitude of direct containment heating. These parameters were selected in accordance with the sensitivity analysis using MAAP 3.0B, recommended by EPRI[5].

As discussed in Subsections 4.6.1.1 through 4.6.1.7, 16 KPDSs were grouped into seven categories based on their similarities (page 4.6-2) in the presentation of MAAP results:

- RCP seal failures, stuck-open pressurizer PORVs and/or stuck-open pressurizer safety values (KPDSs: HAYDI, HANNI, HANNS; CDF ≐ 61.5 percent)

- Loss of both secondary heat removal and containment heat removal (KPDSs: SXNNS, SXNNI, SXNNL; CDF = 6.9 percent)

- Loss of secondary heat removal; RWST injected into containment and containment heat removal successful (KPDSs: SXYAI, SXYGS, SXYCI, SXYDI; CDF = 15.2 percent)

- Medium LOCAs (KPDSs: INYCI, INNNS; CDF = 8.9 percent)
- Large LOCAs (KPDSs: LNYAI, LNYCI; CDF = 5.7 percent)
- Unisolated steam generator tube ruptures (KPDS: INNGB; CDF = 1.5 percent)
- Interfacing system LOCAs (KPDS: INNGV; CDF = 0.1 percent).

Discussions of the phenomenological uncertainties of severe accident progressions appear in Subsection 4.5.2, page 4.5-2, under Top Events.

2.1.3.2 Dominant Contributors: Consistency with IPE Insights

Table 2 in this technical evaluation report shows a comparison among the conditional probabilities of the various containment failure modes set out in the Diablo Canyon IPE submittal, the Zion IPE submittal, and the Surry NUREG-1150 study[3, 6]. The Diablo Canyon results are shown in columns 4 and 5 of the table. Column 4 reflects the IPE results without taking credit for recovery of containment heat removal 48 hours after event initiation. Column 5 shows the IPE results after taking such credit.

Table 2 Containment Failure As a Percentage of Total CDF for Internal Initiators: Comparisons to Zion IPE and Surry NUREG-1150 PRA Results [3, 6]

Containment Failure	Surry/NUREG-1150	Zion IPE	Diablo Canyon[2]	Diablo Canyon[3]
CDF (per reactor year)	4.1E-5	4.0E-6	8.8E-5	8.8E-5
Early	1	0	4.6	4.6
Late	6	5	66.6	45.2
Bypass	12	30	1.8	1.8
Isolation	+	2	7	7
Intact	81	63	20	41.4

+ Included in early failure
2 Reflects the IPE results without taking credit for recovery of containment heat removal 48 hours after event initiation
3 Reflects the IPE results after taking credit for recovery of containment heat removal 48 hours after event initiation

The Diablo Canyon IPE treats the early containment failure in terms of two release category groups: Small, early containment failure; and large, early containment failure. The Diablo Canyon IPE also considers the containment isolation failures as part of a small, early containment failure. For the purpose of comparing the Diablo Canyon IPE results to those of the Zion and Surry studies, SCIENTECH combined the early release category groups into one and removed the contribution of containment isolation failures from the combined group. According to Subsection 4.8 of the submittal, the percentile contributions to CDF of small and large release categories were 8.7 and 2.9, respectively. The 8.7 percentile contribution of the small release category includes that of containment isolation failures. Based on Subsection 4.8.2.1 of the submittal, 80 percent of the sequences leading to small, early containment failures were related to containment isolation failures. Based on these data, the percentile contribution of CDF to containment isolation failure might be roughly 7 percent (80 x 8.7%). The authors of the Diablo Canyon IPE submittal claim that the containment isolation failures were treated conservatively in the Level 1 analysis.

The major causes of late containment releases are threefold: long-term overpressurization (73 percent), basemat melt-through (19 percent), and hydrogen burns resulting in large failures. The IPE submittal claims that many long-term pressurization sequences may be followed by a 90-percent probability of recovery (page 4.8-5 of the submittal).

Containment bypass release groups include sequences involving both interfacing system LOCAs and steam generator tube ruptures. The percentile contribution of these types of accidents to core damage frequency was found to be substantially lower in the Diablo Canyon IPE as compared with what was found at Zion or Surry.

The Containment Intact category comprises approximately 20 percent of the core damage frequency, taking no credit for recovery of containment heat removal. If credit is taken for this recovery action at Diablo Canyon, the long-term containment release group percentage becomes approximately 41 percent, which is still lower than Zion's.

2.1.3.3 Characterization of Containment Performance

As noted in Subsection 4.5, page 4.5-2:

> Phenomenological questions were admitted as CET top events if they addressed containment failure events and failure modes.

Many CET top events addressed containment performance. Containment loadings were obtained from MAAP simulations (Subsection 4.6.1, page 4.6-1). In summary, the IPE appropriately characterized containment performance for each of the CET end-states by assessing containment loading.

In Subsection 3.1.6.2 on page 3.1-98, it is stated that:

> Plant walk-throughs determined only a limited amount of water will be in the reactor cavity prior to melt-through, even if the RWST is injected.

This observation led the IPE submittal authors to categorize the vessel failure behavior into two types:

1. Reactor vessel failures at low RCS pressure
2. Reactor vessel failures at high RCS pressure

In the case of Type 1, characterized as "low pressure melt ejection," it has been argued that core-concrete interaction is the primary reaction. The reactor cavity is in effect dry before the vessel failure. The water in the RCS introduced to the reactor cavity at the time of vessel breach is considered insignificant. Interaction between the debris and the water is regarded as unlikely.

In the case of Type 2, the submittal notes on page 4.1-7 that:

> The high pressure reactor vessel failures are associated with loss of secondary cooling events, small LOCAs and some of the smaller RCP seal failures. When the vessel fails, the core debris is ejected from the vessel with large forces caused by the high pressures in the reactor vessel. These explosive forces would very likely force open the RCDT Room door, the dampers and the access covers. Water would flood the cavity at this time.

2.1.3.4 Impact on Equipment Behavior

A discussion of the impact of severe accidents on equipment is presented in Subsection 4.6.2 of the submittal. The following major points were made with regard to equipment survivability:

- Piping, electrical power cables, and control cables at penetrations are likely to maintain functionality.

- Compartments most likely to have high, local hydrogen concentrations are the cavity and the lower compartment, which contains little equipment required for accident mitigation.

- Containment fan cooler units (CFCUs) and containment spray headers are located in the largest and best-mixed compartment–the upper compartment. Containment heat removal equipment failure due to local hydrogen detonation is unlikely.

- CFCUs are qualified and tested for temperatures up to 324°F and for 100-percent humidity. They retain component capability at temperatures greater than 400°F,

which is adequate for all sequences throughout the accident progression. The total integrated radiation dose is expected to be within qualification limits.

- Containment spray pumps and MOVs would not be affected because they are located outside the containment. Damage as a result of alpha mode vessel failure to the spray header riser or rings is possible, although unlikely.

- ECCS pumps are located outside containment and would not be affected by severe accident conditions.

2.1.4 Reducing Probability of Core Damage or Fission Product Release

2.1.4.1 Definition of Vulnerability

In the last paragraph in Section 7 of the IPE submittal, the authors state that "no containment performance vulnerabilities were identified in this study."

Subsection 4.8.3 of the submittal identifies containment bypass sequences (i.e., interfacing system LOCAs and unisolated SGTRs) to be of special concern because they lead to an early source term release. Citing NUMARC Report 91-04, "Severe Accident Issue Closure Guidelines," the IPE submittal states that:

> The frequency of interfacing system LOCAs and induced SGTRs sequences are approximately 1.0E-7 per year. The NUMARC closure guideline states that no action is necessary . . .

Two containment bypass core damage accidents that require action based on NUMARC recommendations are characterized by steam generator tube ruptures caused by the following:

- Stuck-open steam generator atmospheric dump valves or a safety relief valve with a core damage frequency of 8.7E-7 per year

- Operator failure to terminate ECCS injection (overfill) with a core damage frequency of 4.3E-7 per year.

The IPE submittal states that Diablo Canyon will consider the insights gained from analyzing these two accidents when establishing DCPP Severe Accident Management Guidelines. The submittal further states that no other containment bypass sequences were identified that would require taking a recommended action.

2.1.4.2 Plant Improvements

There are several references to a scheduled plan to remove the HEPA filters in the next refueling outage. This action is apparently related to DCPP concern about the

survivability of CFCUs during a severe accident . The IPE authors argue that the aerosols could plug the HEPA filters. No explicit statement affirms that this action was taken as a result of the IPE study.

2.1.5 Responses to CPI Program Recommendations

One recommendation of the CPI program pertaining to PWRs with large, dry containments is that utilities evaluate containment and equipment vulnerabilities to hydrogen combustion (local and global) as a part of IPE analysis and identify needs for improvements in procedures and equipment. The Diablo Canyon response to this recommendation appears in Subsection 4.6.2, paragraph 3, page 4.6-49, of the IPE submittal, where it is argued that containment and equipment vulnerabilities to hydrogen combustion are unlikely:

> It has been concluded that containment or containment heat removal equipment failure due to local hydrogen detonation is unlikely.

However, no discussion can be found in response to CPI recommendations on global hydrogen combustion.

2.2 IPE Strengths and Weaknesses

2.2.1 IPE Strengths

1. The back-end portion of the IPE benefitted from a well-documented and reviewed Level 1 PRA, performed as a part of the Long-term Seismic Program.

2. The IPE submittal is well written and appears to be technically sound.

3. The IPE submittal report uses the exact format given in NUREG-1335, which makes the report easy to follow.

2.2.2 IPE Weaknesses

1. The quantification process of the CET is not discussed.

2. Although the IPE cites as an objective the identification of differences between the Level 2 results at Zion and DCPP, no attempt was made to gain any insight by comparing these results.

3. The IPE results indicate that the containment is vulnerable to isolation failures. In the discussion of the containment performance vulnerability, no reference is made to this particular failure mode.

2.3 Questions, Comments, and Requests for Further Information on the Diablo Canyon IPE Back-end Submittal

1. In Section 4.1.1.5, page 4.1-3, of the IPE submittal, the possibility is discussed of high-efficiency particulate air (HEPA) filter plugging due to suspended aerosols in the containment atmosphere. Reference is made to a planned design change to remove HEPA filters. What are the safety implications associated with this planned design change?

2. The failure mode descriptions, as given in Table 4.4-1, agree with the text on page 4.4-2. However, in Table 4.4-2, 14 failure modes (A through N) are listed for which there is no supporting discussion. What are the relationships between these two sets of failure modes?

3. As stated in Subsection 4.7.2.3.1, paragraph 4, page 4.7-7:

   > For all MAPP calculations, the reported source terms are the release fractions at 50 hours (approximately 2 days) after initiation of the event .... [S]imulations which were allowed to run for long periods did not show huge increases in the release fractions over the release fraction at 50 hours.

   What was the maximum of such an increase in source terms?

4. Seven areas of uncertainty associated with direct containment heating are listed in Subsection 4.5.2, page 4.5-6, of the submittal. Please explain how these uncertainties are reflected in CET split fractions.

5. Subsection 4.6.1.2.1, page 4.6-13, of the IPE submittal states:

   > When the blockage model is used, it has the effect of reducing gas circulation through the core as the core degrades.

   What is the typical reduction of such gas circulation?

6. If induced RCS hot leg or surge line failure (Top Event IP of CET) occurs, what is the assumed likelihood of in-vessel cooling of debris by coolant injection?

7. Subsection 4.6.2, paragraph 3, page 4.6-49, of the IPE submittal responds to CPI Program recommendations. What analyses were performed in this regard?

8. The IPE takes credit for many long-term pressurization sequences. The probability of this recovery action is noted as 90 percent. What is the basis for this probability value? Does this probability apply to both CFCUs and sprays?

9. A discrepancy appears in the IPE submittal with respect to the impact of CFCU recovery in preventing late containment failure. In one place in the submittal, the contribution of CFCUs in arresting late containment failure was assessed as marginal, as shown in Table 4.8-2. Earlier, in Subsection 4.8.2.5, it could be inferred that, by taking credit for this recovery, a 20-percent reduction in late containment failures could occur. Please explain this apparent discrepancy.

10. Given the susceptibility of the containment to isolation failure, why doesn't the DCPP take any action to reduce the likelihood of this event? If the isolation failure is treated conservatively in the IPE, what is your best estimate for containment isolation failure?

11. As noted in Subsection 4.8.4.3, page 4.8-10, of the IPE submittal, a relatively simple design modification would allow cavity flooding. Why doesn't DCPP consider this design modification in light of the sensitivity analysis results, which show a 50-percent reduction in large, early containment failures?

12. Are there any threats to containment integrity caused by recovering the CFCUs, thereby de-inerting the containment?

13. For induced large LOCAs, described on page 4.6-20, you postulate a large vessel failure due to pressurized thermal shock at the time of RWST water injection. For this scenario, you claim that the availability of recirculation water and CFCUs ". . . would prevent this fuel damage accident from evolving into a severe accident . . . ." Please provide the analysis that supports this claim.

3. OVERALL EVALUATION AND CONCLUSIONS

As discussed in Section 2, this IPE submittal appears to contain sufficient back-end information to address the severe accident vulnerability issues at Diablo Canyon. Once the weaknesses listed in Subsection 2.2 and the questions listed in Subsection 2.3 are addressed by PG&E, we believe the IPE will be completely responsive to the intent of the GL and the IPE program to resolve severe accident issues at Diablo Canyon. The back-end portion of the submittal is thorough and well written.

The major factor that drives early containment failure is the failures of containment isolation. According to Level 2 sensitivity analyses, as illustrated in Table 4.8-2 of the submittal, elimination of these types of failures leads to an 80-percent reduction in early containment failure. Reduction of isolation failure may be worth pursuing.

Another way to reduce an early and large containment failure may be to prevent vessel bottom head failure by flooding the reactor cavity. As indicated in Table 4.8-2 of the submittal, this action reduces early containment failure by 50 percent and late containment failure by 19 percent. However, flooding would require modification of the RCDT room door. PG&E expressed some concern that flooding the reactor cavity might increase the likelihood of ex-vessel steam explosion. Nevertheless, this modification might be worth exploring.

4. REFERENCES

1. "Diablo Canyon Power Plant Units 1 and 2 Individual Plant Examination Report," prepared by Pacific Gas and Electric Company, April 1992.

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

3. "Severe Accident Risk: An Assessment of Five U.S. Nuclear Power Plants," NUREG-1150, 1990.

4. "Evaluation of Severe Accident Risks: Zion Unit 1," U.S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 7, October 1990.

5. EPRI, "Recommended Sensitivity Analysis for An Individual Plant Examination Using MAAP 3.0B," EPRI Report TR-100167.

6. "Evaluation of Severe Accident Risks: Surry Unit 1," U.S. Nuclear Regulatory Commission, NUREG-4551, Vol. 3, Part 1, June 1990.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-End Facts

Plant Name

Diablo Canyon

Containment Type

Large, dry containment

Unique Containment Features

Water collected on the containment floor cannot access the reactor cavity

Unique Vessel Features

None found

Number of Plant Damage States

16

Ultimate Containment Failure Pressure

120 psig (median); this value reflects the weakest point in the containment corresponding to failure modes I, J, and K, as shown in Table 4.4-2 of the submittal

Additional Radionuclide Transport And Retention Structures

Auxiliary Building retention credited for the interfacing system LOCA sequences

Conditional Probability That The Containment Is Not Isolated

0.07